Appendix C

Topical Report
Residual Radionuclide Distribution and Inventory at the Pathfinder Generating Plant

PNL-4326

PATHFINDER HEARING FILE, ITEM NO. 1

TOPICAL REPORT

RESIDUAL RADIONUCLIDE DISTRIBUTION AND INVENTORY AT THE PATHFINDER GENERATING PLANT

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June, 1982

Prepared for
the U. S. Nuclear Regulatory Commission
under a Related Services Agreement with the
U. S. Department of Energy
Contract DE-AC06-76-RLO 1830, FIN No. B2299

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1.0 INTRODUCTION

The U. S. Nuclear Regulatory Commission (NRC) has been charged with the responsibility of developing a general decommissioning policy for nuclear facilities in the United States, including nuclear power plants. Since the nuclear industry has matured to the point where some of the early nuclear power plants have reached retirement status, it is imperative that the NRC develop detailed information to provide guidance for the decommissioning of these plants.

Several studies have conceptually assessed the technology, safety, and costs associated with various alternatives for decommissioning nuclear power plants.[1-5] One of the key elements of such assessments is a characterization of the radionuclide inventory within a retired nuclear power plant. This information is essential for understanding the radiological problems which will be encountered during decommissioning. However, empirical data relating to the composition, distribution, and quantity of residual radionuclides within nuclear power plants are extremely meager, and calculated or estimated quantities have mainly been used in previous assessments.

To provide the NRC with an actual data base of residual radionuclide measurements within nuclear power plants, Pacific Northwest Laboratory (PNL) has contracted to conduct a comprehensive sampling and analyses program at a number of nuclear power plants. The objective of these studies is to provide information on the range of types, quantities, and locations of radionuclide residues likely to be encountered in retired reactor power stations (exclusive of the reactor pressure vessel) and in the immediate station environs. This program was initiated at the shutdown Pathfinder Generating Plant and includes planned measurements at two more retired nuclear power plants and several operating stations.

The on-site sampling and measurements program was conducted at the Pathfinder Generating Plant in July, 1980, to determine the residual radionuclide concentrations, distribution, and inventory at the plant. The program emphasized the characterization of radionuclides which had been transported from the pressure vessel and deposited in the various systems and components of the plant. Samples of soil surrounding the plant were also analyzed for residual radioactive material. The results of the sampling and measurement program for the Pathfinder Generating Plant are summarized in this report.

2.0 SAMPLING AND MEASUREMENTS PROGRAM

The sampling and field measurements at Pathfinder were conducted in July, 1980. A portable intrinsic Ge gamma-ray spectrometer and a beta-gamma detector were set up in the water chemistry laboratory.

The intrinsic Ge detector (a 30% efficient Princeton Gamma-Tech coaxial diode with a resolution of 1.94 keV FWHM) was housed in a lead brick shield. The detector signal output was coupled to a Canberra Series 80 multichannel analyzer and minicomputer (see Figure 2.1). A magnetic tape deck and a hard copy printer were also interfaced to the multichannel analyzer to provide on-site data storage or printout of the reduced data. This detector and counting system was designed to be highly portable and was also used to determine in-situ radionuclide concentrations on concrete floors and in intact piping (see Figures 2.2 and 2.3).

The beta-gamma detector (Eberline) was used primarily for monitoring smearable contamination levels on floors and equipment.

2.1 SOIL SAMPLING

A survey of the soils immediately surrounding the power plant was conducted using a G-M counter in an unsuccessful attempt to locate contaminated soils. Since no detectable contamination could be observed with the G-M, soil samples were collected at ten selected locations around the power plant (see Figure 2.4). A special large-area coring device was used to obtain soil cores 450 cm^2 in area to a depth of 6 cm (see Figures 2.5 and 2.6). Cores were taken at sites S-1, S-2, S-3, S-4, S-7, and S-9 (see Figure 2.7). Soil Sample S-5 was a grab sample removed by shovel from the drainage ditch which was formerly used for discharging low-level aqueous rad-wastes to the Big Sioux River (see Figure 2.8). Soil Sample S-6 consisted of a grab sample of the top 5 cm of soil from the water treatment effluent basin. Samples S-8 and S-10 were collected in areas partially covered with crushed rock and were composite samples of the top 2 cm of soil. All soil samples were sieved through No. 20 mesh screens and packaged in polyethylene bags. Subsamples of

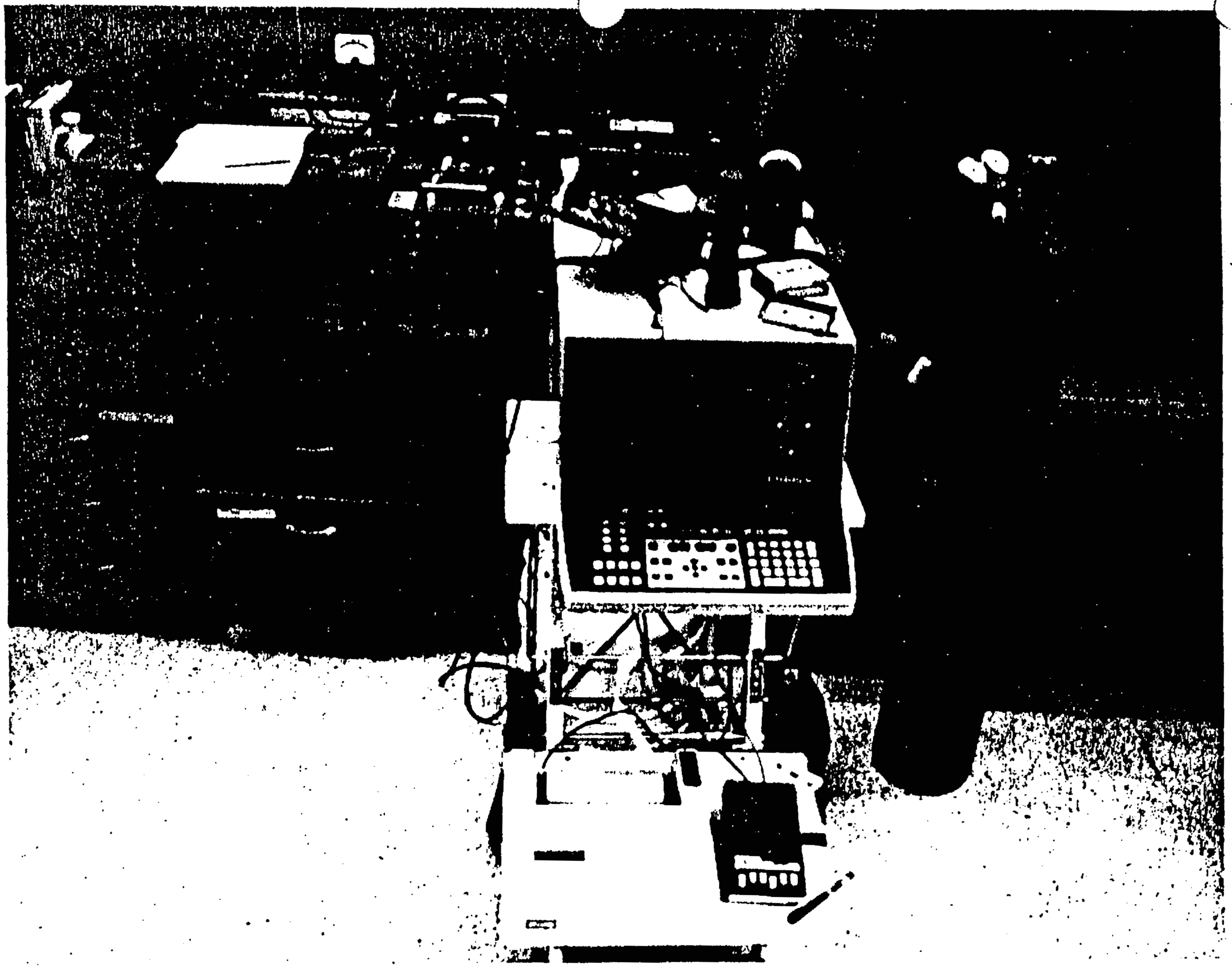


FIGURE 2.1. Intrinsic Germanium Gamma-Ray Spectrometer Set-Up in Chemistry Lab at Pathfinder

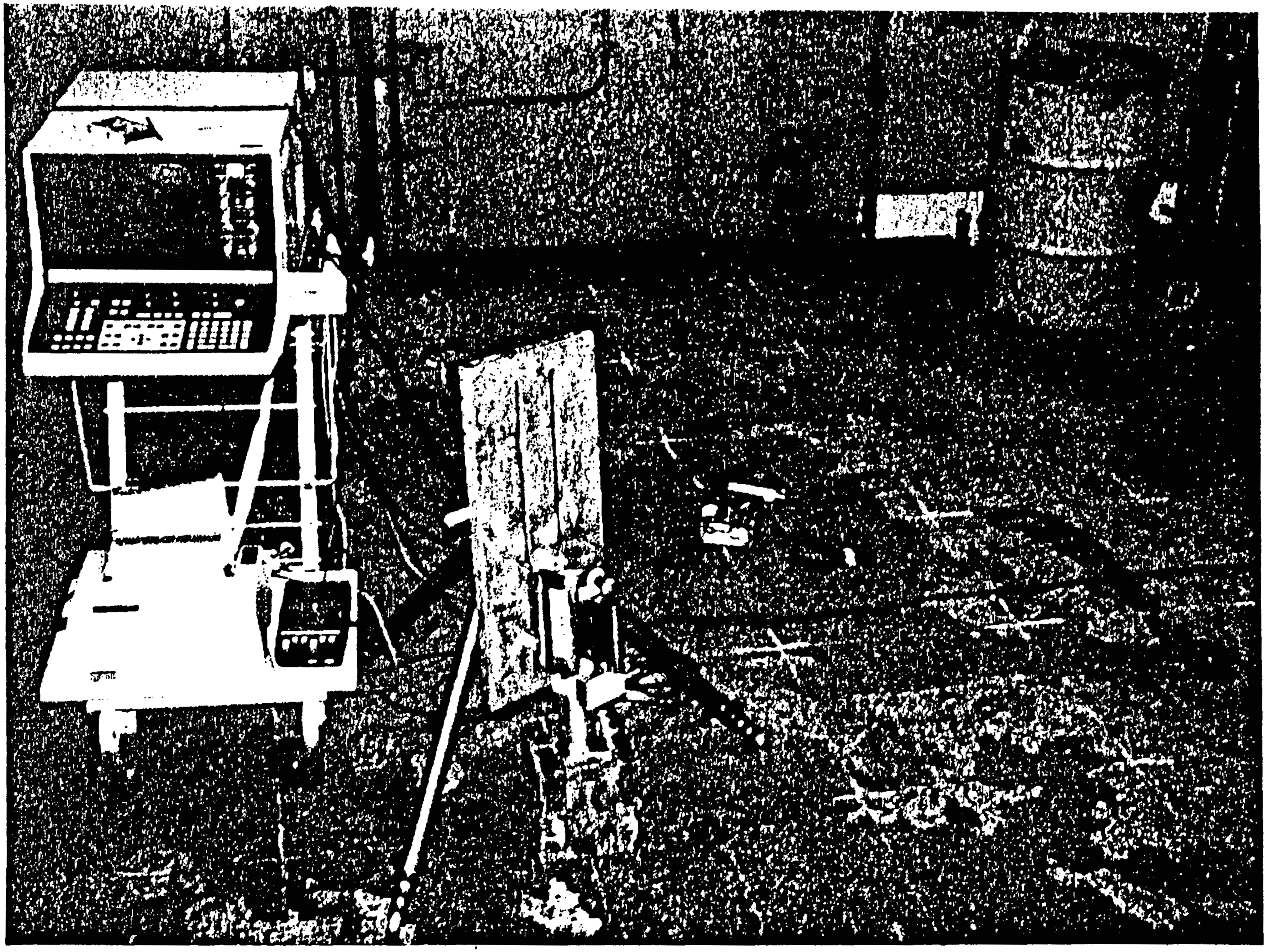


FIGURE 2.2. Portable IG Gamma-Ray Spectrometer During In-Situ Analysis of Concrete Floor

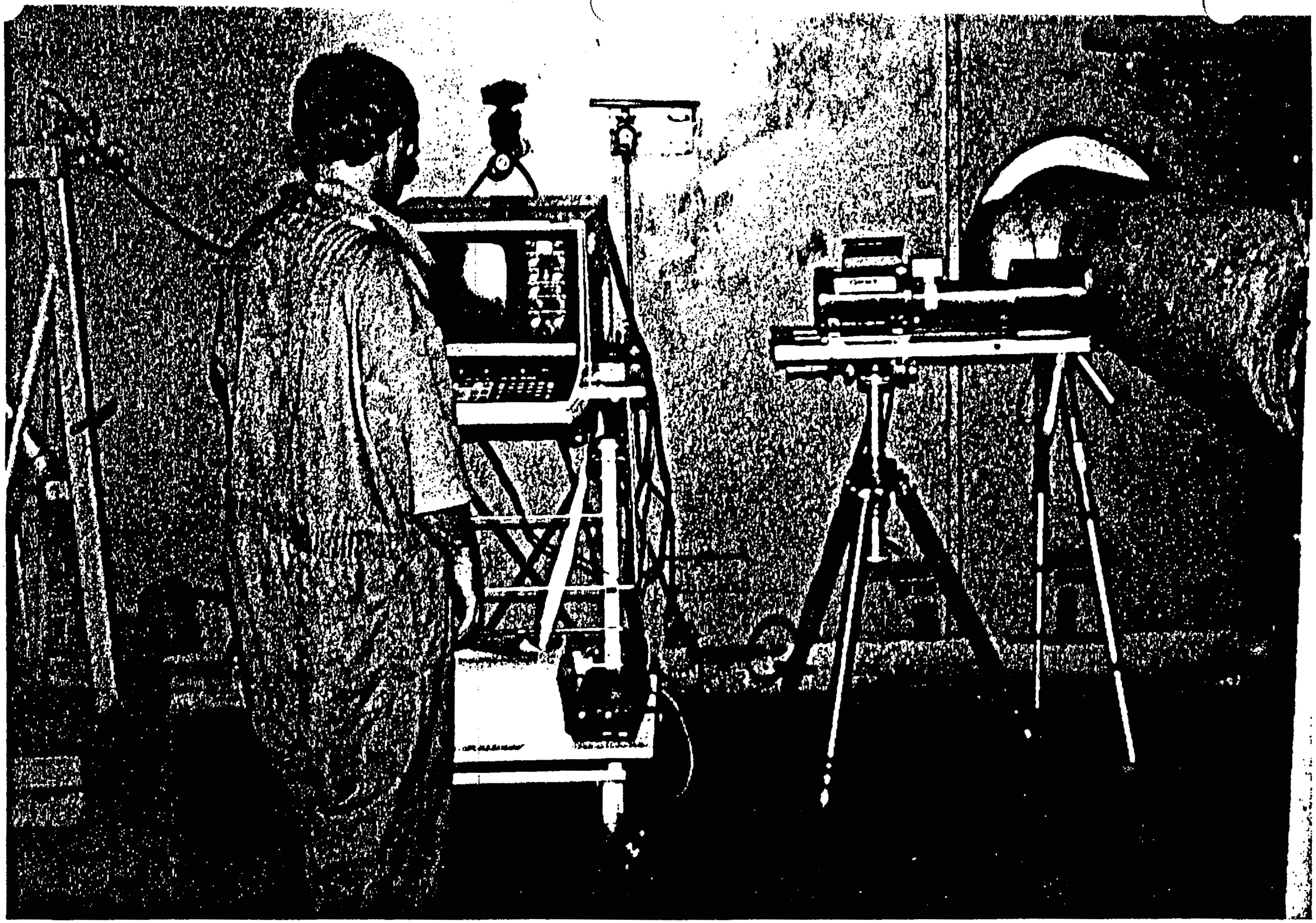


FIGURE 2.3. Portable IG Gamma-Ray Spectrometer During Nondestructive Analysis of Condensate Pipe

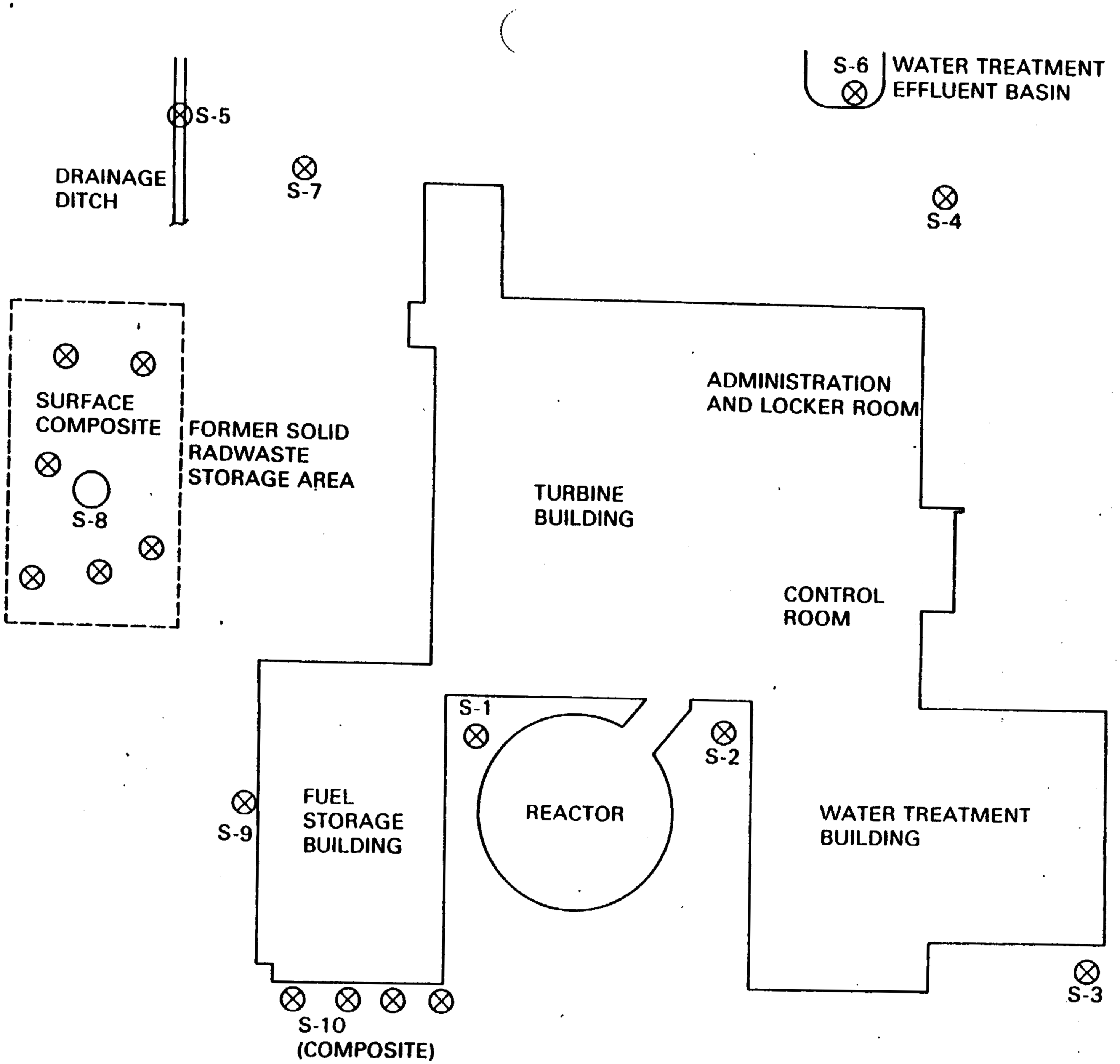
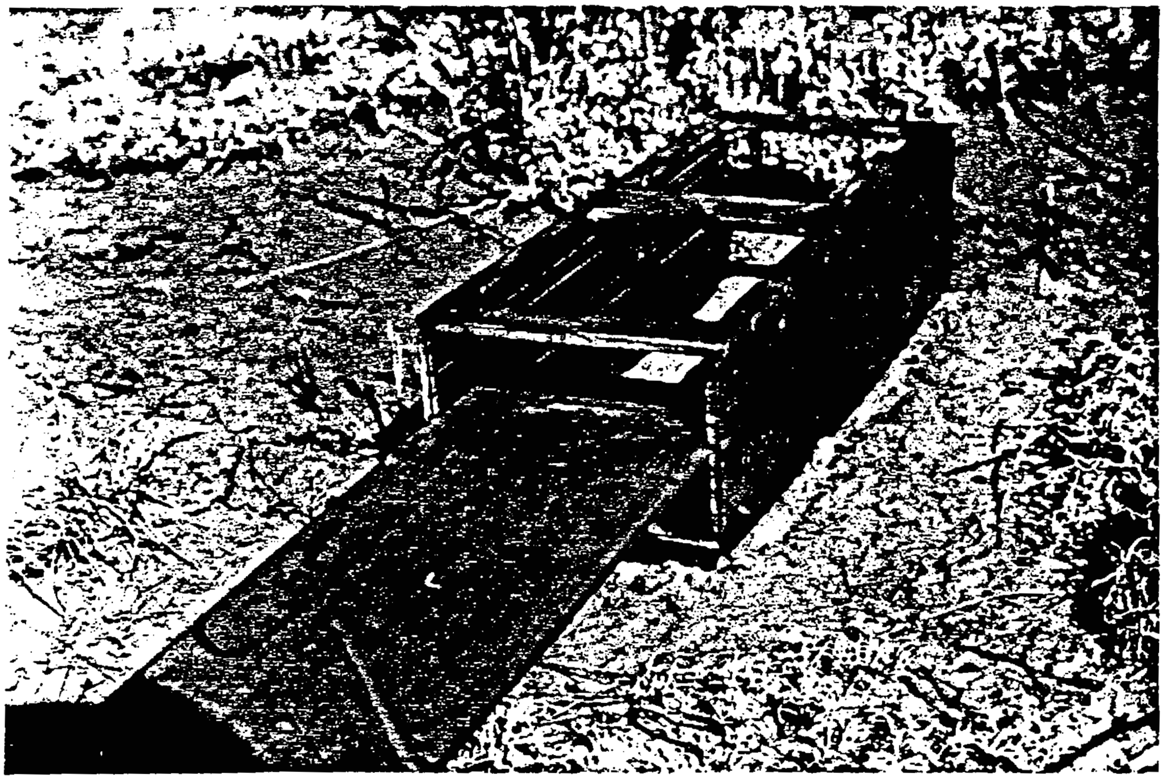


FIGURE 2.4. Soil Sampling Locations Around Pathfinder Plant



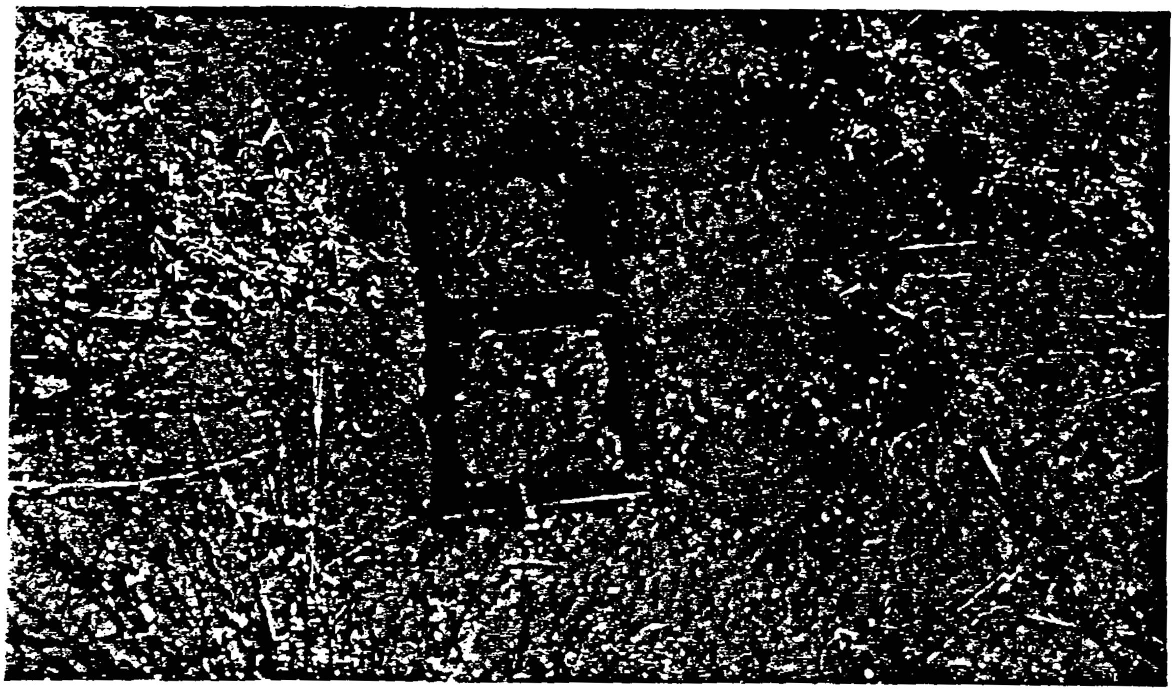


FIGURE 2.5. Soil Corer for Sampling Top 5 cm of Soil at 1-cm Intervals. Top photo shows corer placed on soil. Bottom photo shows corer driven into soil

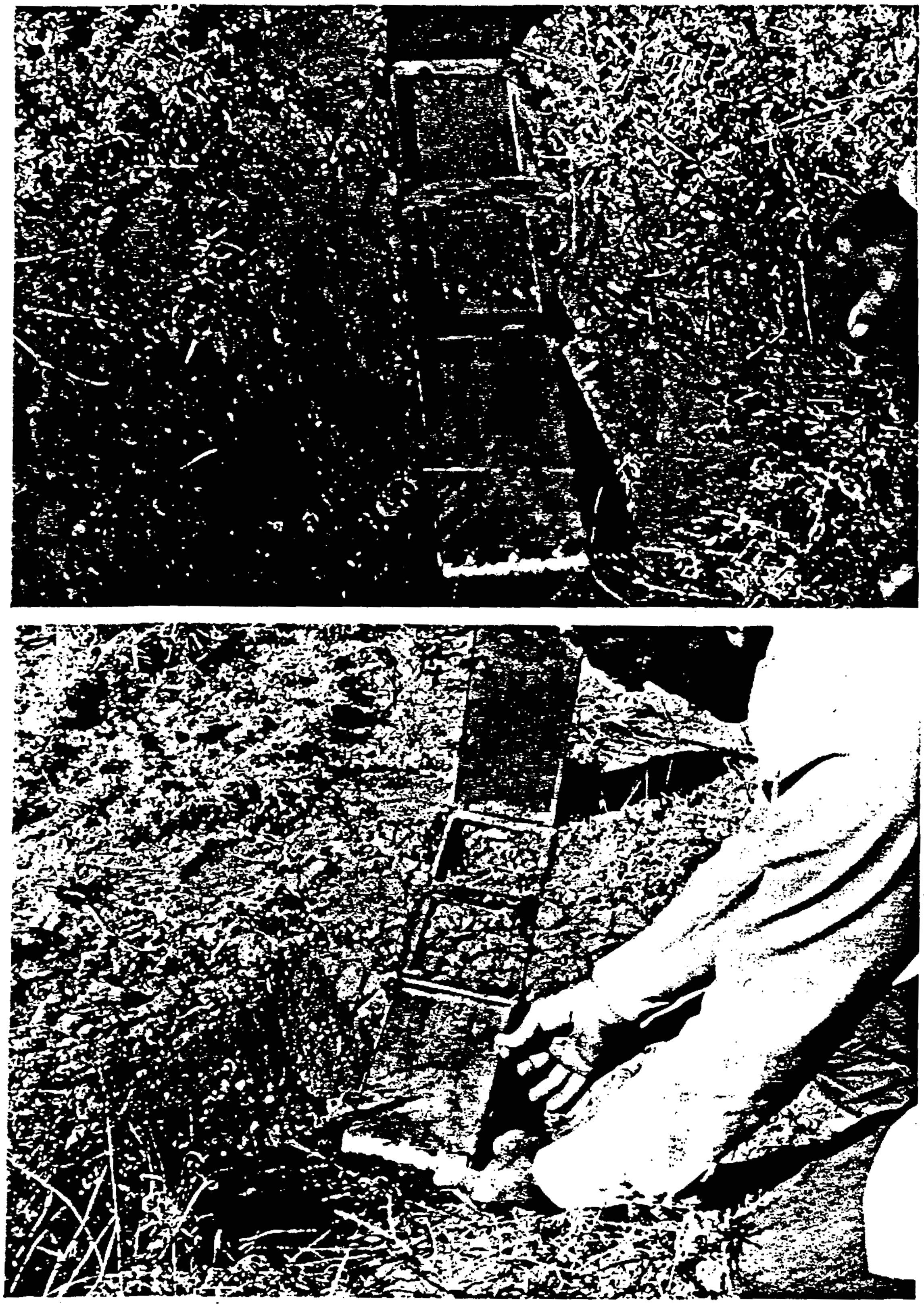


FIGURE 2.6. Soil Corer for Sampling Top 5 cm of Soil. Top photo shows collection of surface layer (top cm) of soil. Bottom photo shows collection of subsurface layer of soil

FIGURE 2.7. Soil Coring at Pathfinder

FIGURE 2.8. Grab Sampling Sediments from Former Rad-Waste Effluent Ditch at Pathfinder

soil (nominally 250 g) were measured onsite by gamma-ray spectrometry to identify any unusually contaminated area which might require more extensive sampling. However, no such areas were observed around the Pathfinder Plant. The soil samples were returned to PNL and weighed aliquots of dried (105 °C), sieved soil were packaged in standard counting jars for quantitative Ge(Li) gamma-ray spectrometry. Approximately 200 to 1000 g of soil were used in the radiochemical analyses of the transuranics and beta emitters.

Since no appreciable radioactivity of reactor origin was found in the soils at Pathfinder, it was felt unnecessary to sample vegetation for radionuclide analyses.

2.2 AUXILIARY STRUCTURES

Since the cooling tower at the Pathfinder Plant was contaminated in 1967 with reactor primary system water, the internals of the cooling tower were sampled. Small pieces of wood slats were removed from the baffle sections (see Figures 2.9 and 2.10). Also, sludge from the bottom of the cooling tower was dredged and packaged in polyethylene bags. The wood slats and sludge ere measured for residual radionuclides by Ge(Li) gamma-ray spectrometry.

Other auxiliary buildings at the Pathfinder site include a warehouse, the old administration building (now used as a storage building), and several small instrumentation sheds. These structures were surveyed with a G-M counter and by smearing 100 cm^2 areas. All were found to be free of any detectable radioactive material above background. Since these buildings were reported by NSP to have never been exposed to radioactivity, they were not sampled.

2.3 PIPING AND HARDWARE

During the partial decommissioning and retrofitting of the Pathfinder Plant for fossil-fuel use, piping from a number of key systems was removed and stored on site (see Appendix C). Some of the piping was labeled and stored in the reactor building (see Figure 2.11), and other piping was stored in the fuel storage pool which had been drained and covered with a thick concrete cap. To gain access to the fuel storage pool a five-foot-square opening was

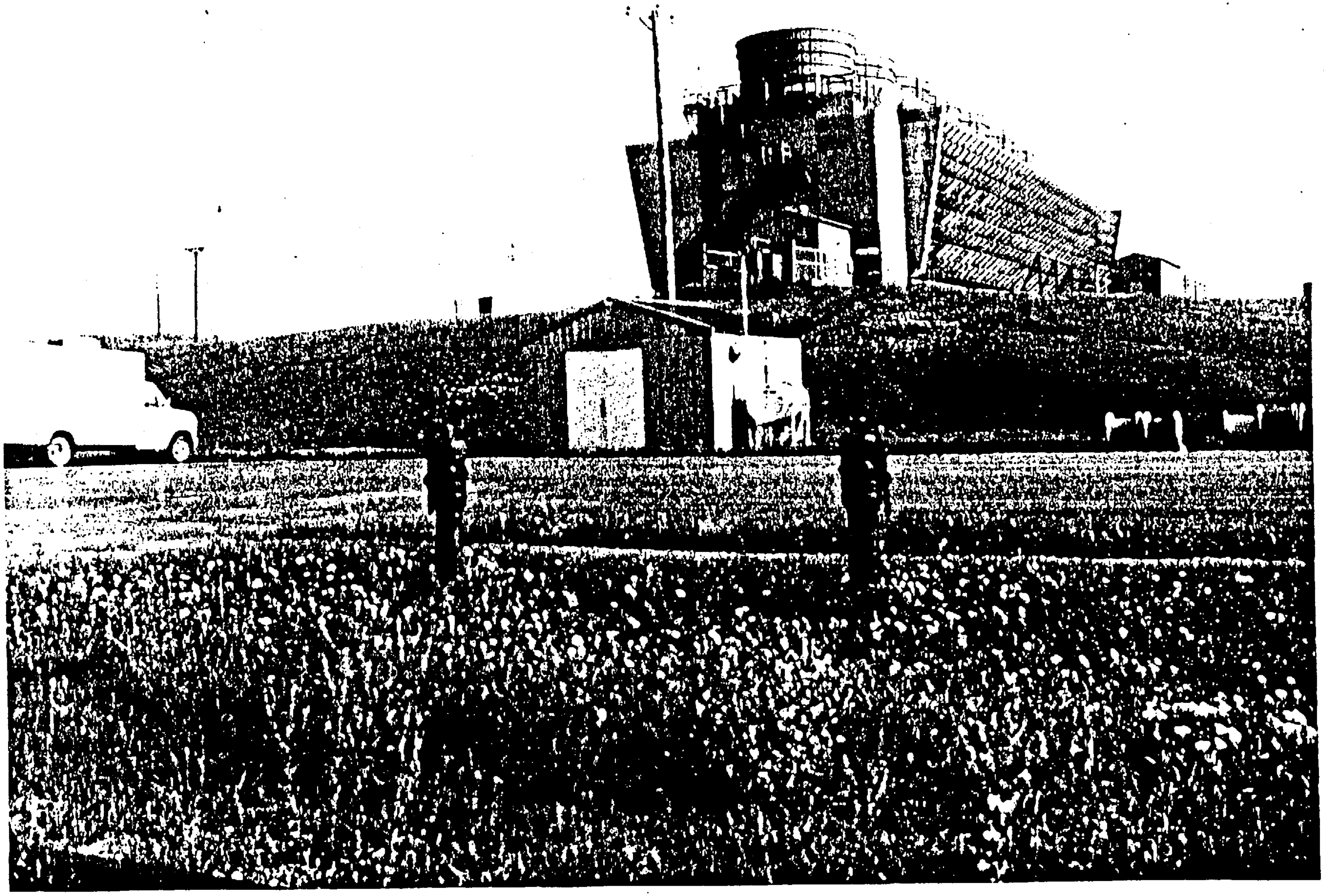


FIGURE 2.9. Cooling Tower at Pathfinder

FIGURE 2.10. Wooden Slats Sampled from Pathfinder Cooling Tower

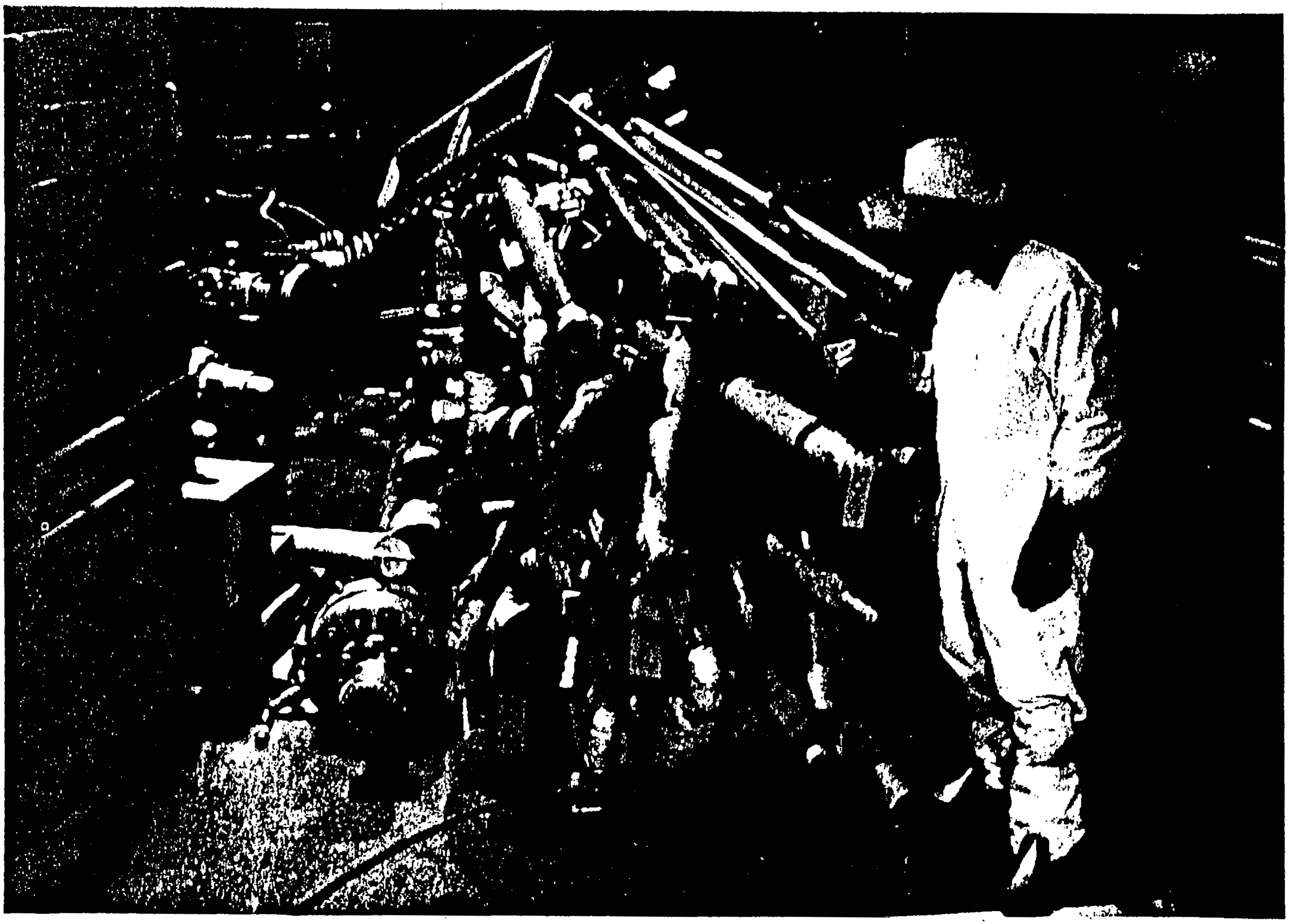


FIGURE 2.11. Piping Samples Removed During Partial Decommissioning at Pathfinder. Stored in reactor building

cut through the concrete cap and a steel framework welded to the exposed rebar (see Figures 2.12 and 2.13). A removable concrete plug was then fabricated for gaining entry into the fuel storage pool. All necessary precautions were taken before entering the fuel storage pool, including measurements of the oxygen and explosive gas concentrations and airborne radioactivity levels (see Figure 2.14). The oxygen concentration was normal and the explosive gas and airborne radionuclide levels were insignificant.

Before entering sealed off radiation zones in the reactor and fuel storage buildings, temporary "greenhouse" structures with appropriate step-off areas to prevent the spread of contamination were constructed at each entry point (see Figure 2.15).

Piping segments (5- to 10-cm dia) stored on the plug floor of the reactor building were sampled by cutting off 15- to 30-cm sections with a portable bandsaw or power hacksaw (see Figure 2.16). Where the pipe diameter was too large for the portable bandsaw, or where piping was to remain intact, a 5-cm-dia core of the pipe was taken using a hole-saw drill (see Figure 2.17). Light use of a cutting fluid was employed to lubricate the saw blade during this operation to prevent overheating of the metal and possible volatilization of radioactivity. Extreme care was used to prevent the spread of contamination during the cutting process. Open ends of piping were taped shut. Diaper paper was spread over the floor under the cutting area, and a large plastic bag was taped under the pipe to catch all of the metal cuttings. The light use of cutting fluid dampened the corrosion film inside the piping at the cutting area just enough to prevent the generation of airborne contamination. During each cut an air sampler was positioned near the pipe to determine if any airborne contamination had been released. In no case was any significant airborne contamination generated, and no contamination of the floors ever resulted. All work areas were smeared after the sampling and were found to be less than 200 dpm smearable.

The same cutting procedures and precautions were followed where piping was still intact within the reactor and fuel handling buildings. The portable bandsaw was used to cut piping up to 10 cm in diameter, and the hole saw was used for coring (5- to 6-cm-dia cores) larger piping (see Figures 2.18,

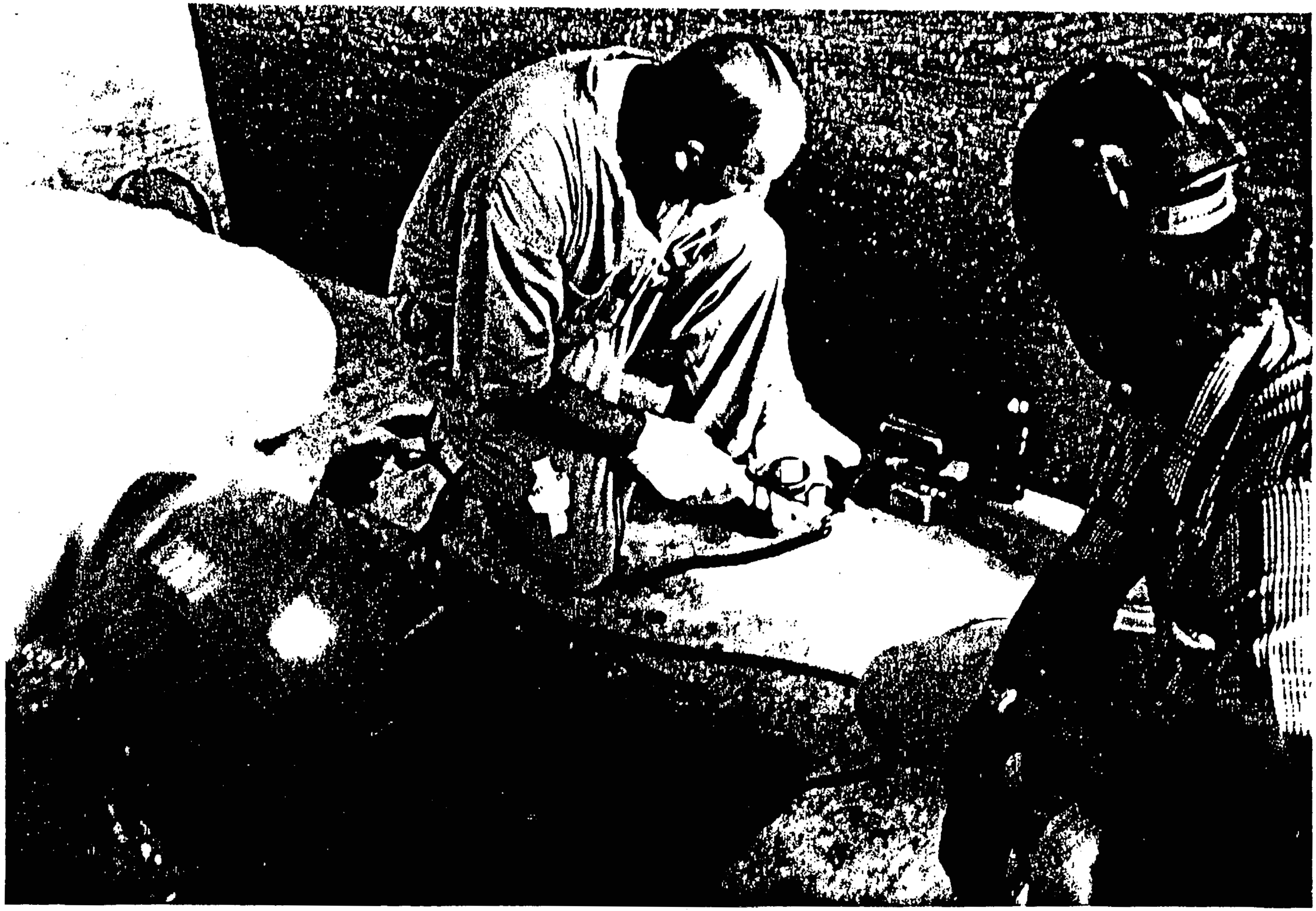


FIGURE 2.12. Entrance Hole Cut Into Fuel Storage Basin. Testing for surface contamination

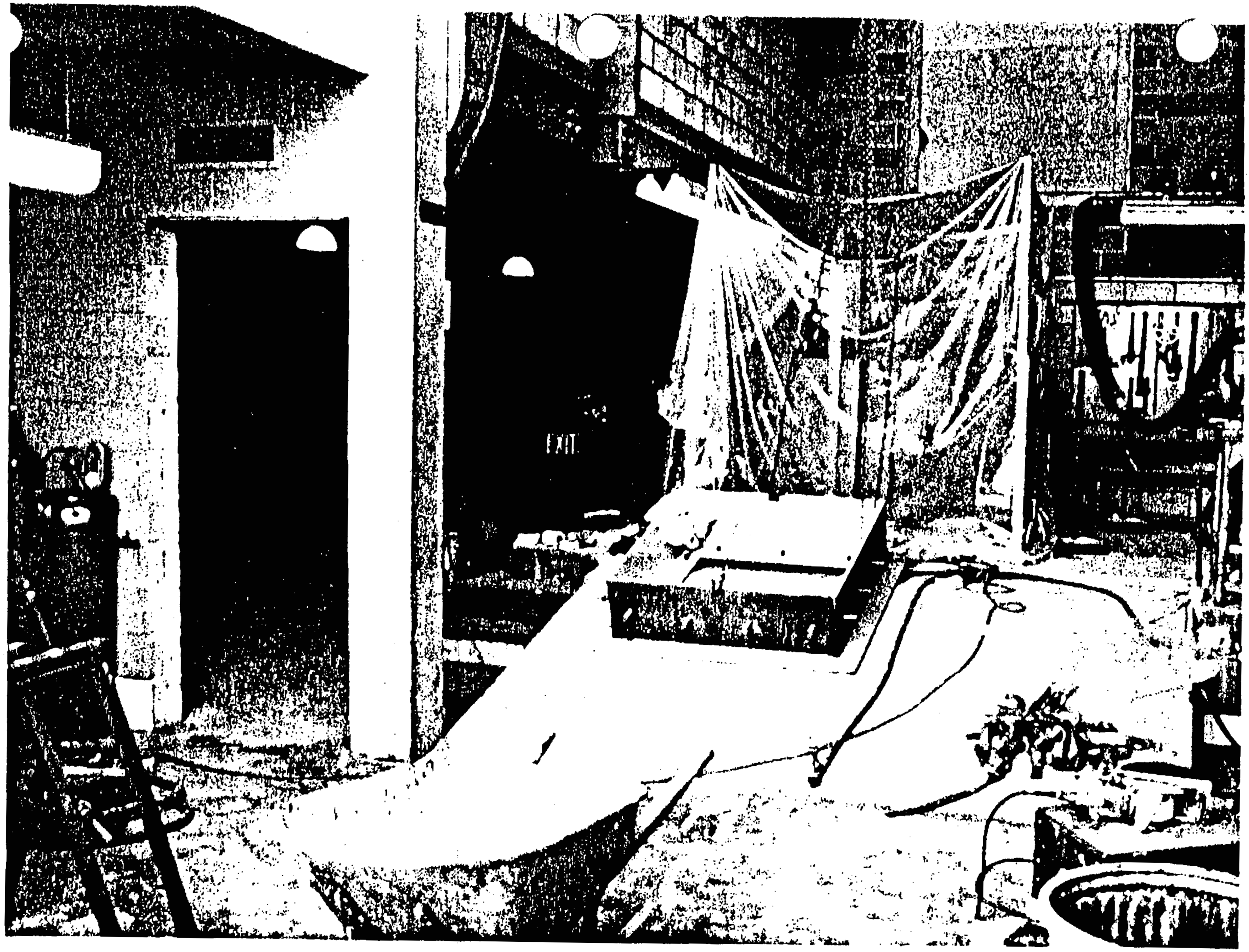


FIGURE 2.13. Steel Framework for Concrete Plug Over Fuel Storage Basin

FIGURE 2.14. Sampling Air, Gases and Radioactivity from Opened Fuel Storage Basin

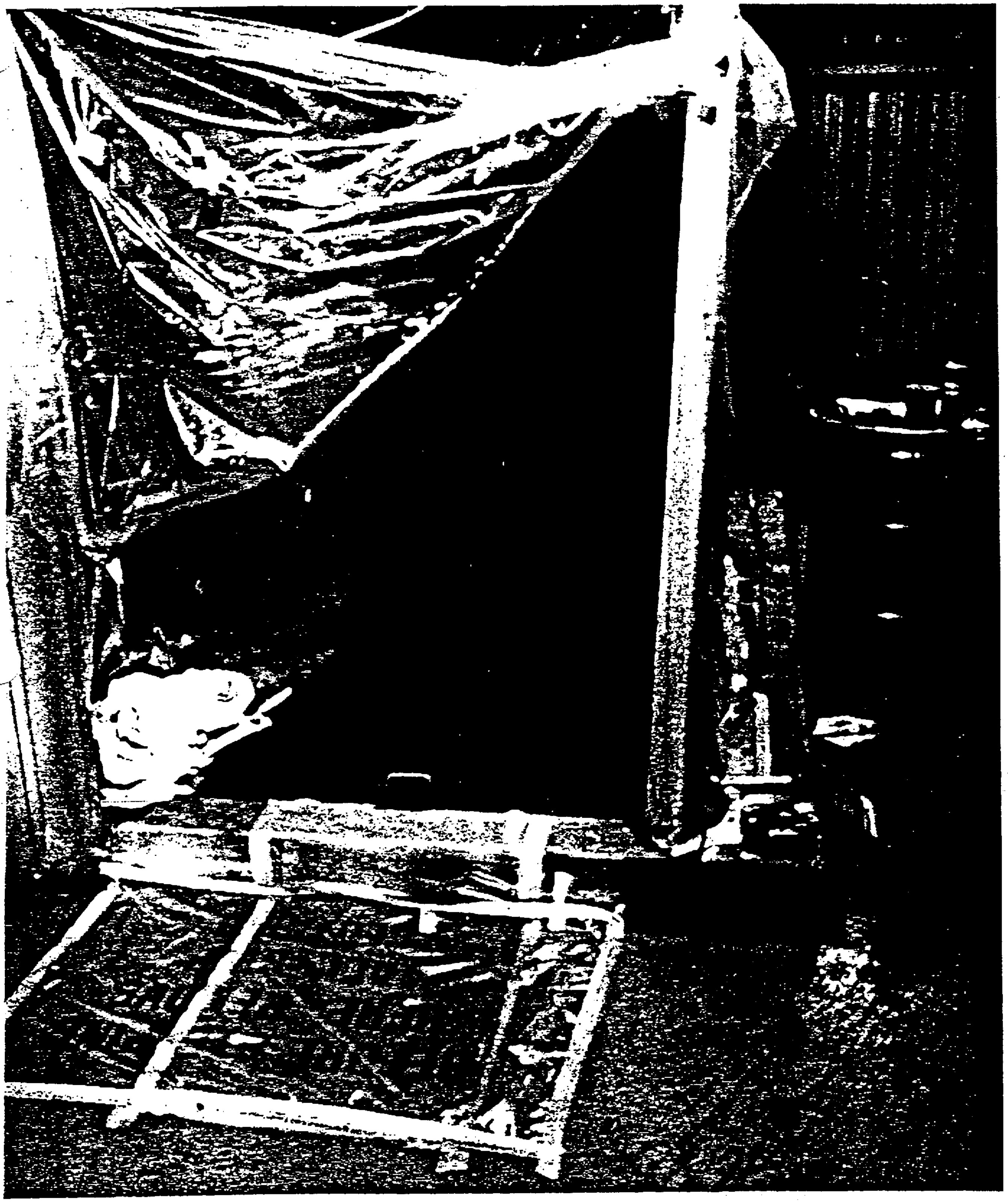


FIGURE 2.15. Greenhouse Structure and Step-Off Pad Installed at Entrance to Lower Floors of Fuel Handling Building

FIGURE 2.16. Cutting Section of Stainless Steel Reactor Water Purification Line With Portable Bandsaw

FIGURE 2.17. Hole-Saw Cutting of Section of Reactor Feedwater Line

2.19, and 2.20. A power hacksaw was also employed to cut out sections of sheet metal from ventilation and air ducts.

All major piping systems associated with the Pathfinder Plant (see Appendix A) which were not presently in use with the fossil-fueled portion of the plant were sampled. These included the main steam line, steam bypass line, reactor feedwater line, purification system line, rad-waste system lines, shield pool clean-up line and off-gas hold up tanks (see Figure 2.21). In addition to the piping systems, pieces of stainless steel hardware were cut from the shield pool and the fuel storage pool for radionuclide measurements. An inventory of the samples collected is given in Appendix D.

All pieces of cut piping and hardware were wrapped in cloth rags to cover jagged edges, double packaged in plastic bags, labeled and finally packaged in metal cans sealed with tape.

2.4 CONCRETE CORE SAMPLING

Twenty-two 10-cm-dia by 12-cm-deep concrete cores were collected from the reactor building, fuel handling building and the turbine building (see Figure 2.22). A Korit® concrete coring apparatus utilizing a diamond-tipped core barrel was employed for the coring work (see Figures 2.23 and 2.24). Water, recirculated through a 15-liter reservoir, served as the drill bit coolant and cutting fluid. Prior to taking a core, the floor was surveyed with a G-M detector to try to locate "hot spots" of contamination. Cores were taken at the "hot spots" when they were found, but in general, the floors at the Pathfinder Plant contained very low levels of contamination and coring sites were usually selected near floor drains.

An attempt was made to core the concrete bioshield surrounding the reactor pressure vessel. Both 2.5-cm and 5-cm dia core barrels were used with the Korit® apparatus. The coring device was attached to the south outside wall of the bioshield on the plug floor level of the reactor building (see Figure 2.25). The concrete bioshield thickness at this point is approximately 3 m. The 2.5-cm-dia core barrel was found to be unsatisfactory because the concrete core would not remain intact within the core barrel and much of this core was lost. The 5-cm-dia core barrel produced a concrete core of

FIGURE 2.18. Cutting Out Section of Reactor Water Purification Line With Portable Bandsaw

FIGURE 2.19. Hole-Saw Cutting of Main Steam Bypass Line

FIGURE 2.20. Sample Specimen (2-in.-dia plug) Removed from Emergency Steam Bypass Line by Hole-Saw Cutting

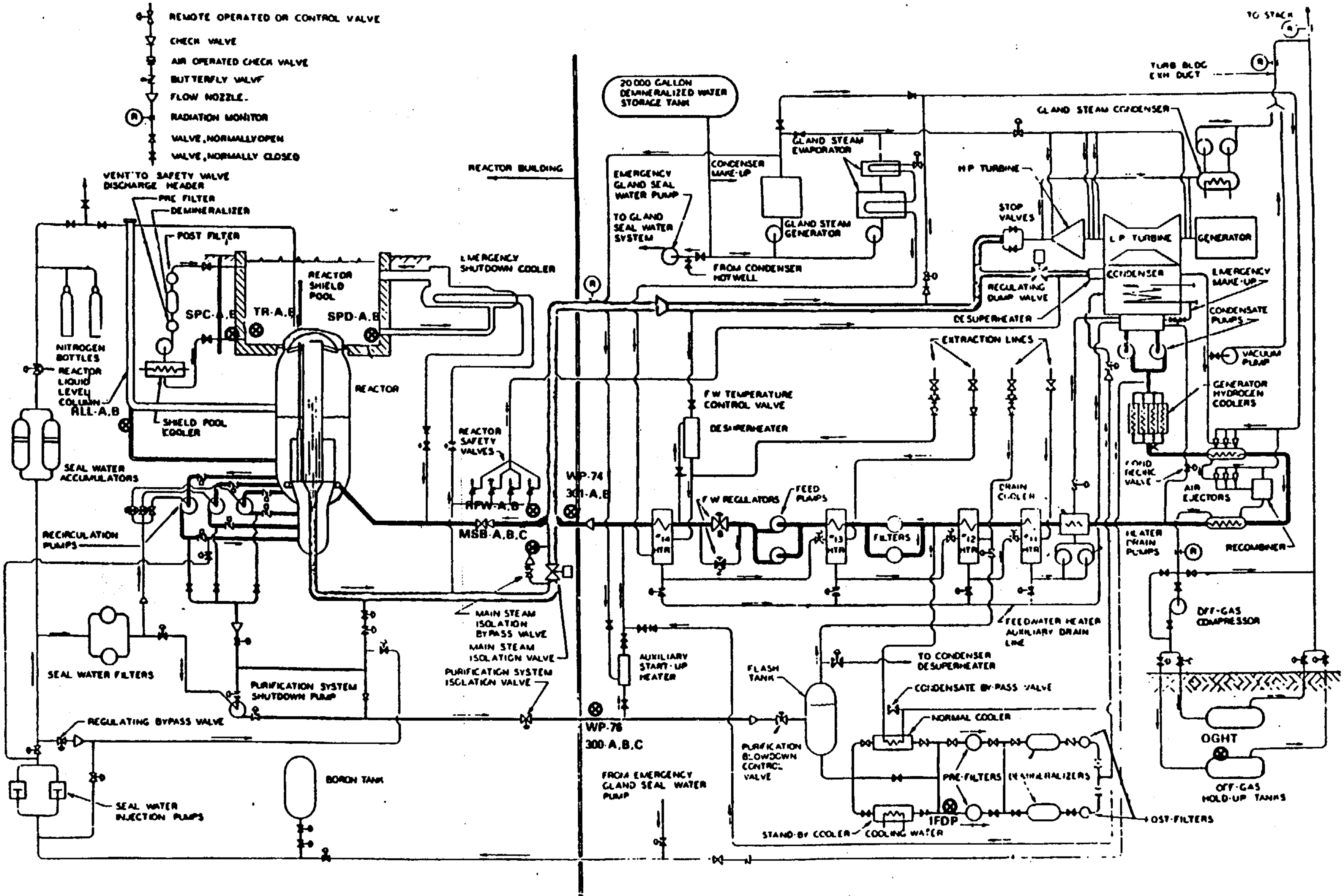


FIGURE 2.21. Schematic Drawing of Plant Systems Showing Sampling Locations of Piping and Hardware

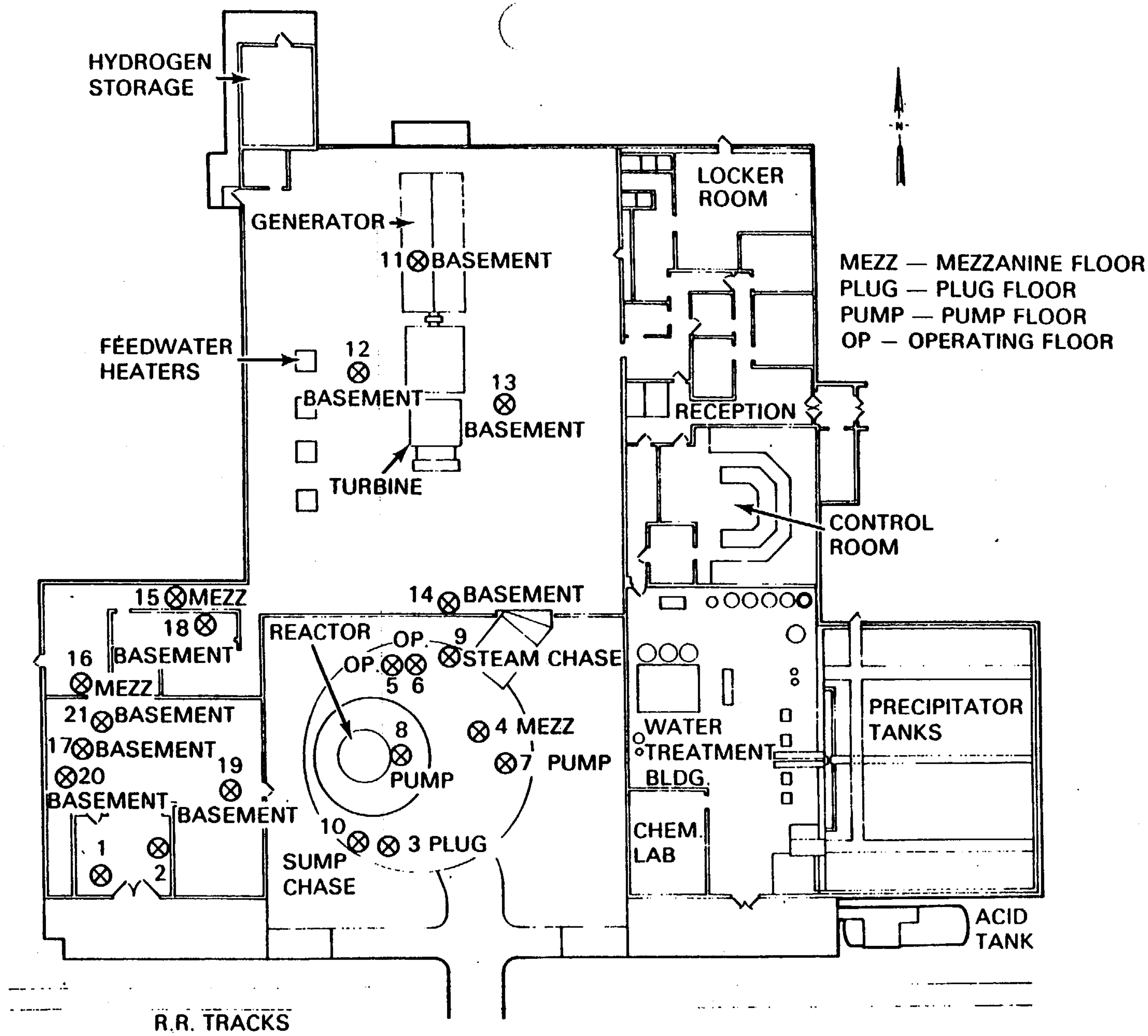


FIGURE 2.22. Concrete Core Sampling Locations at Pathfinder

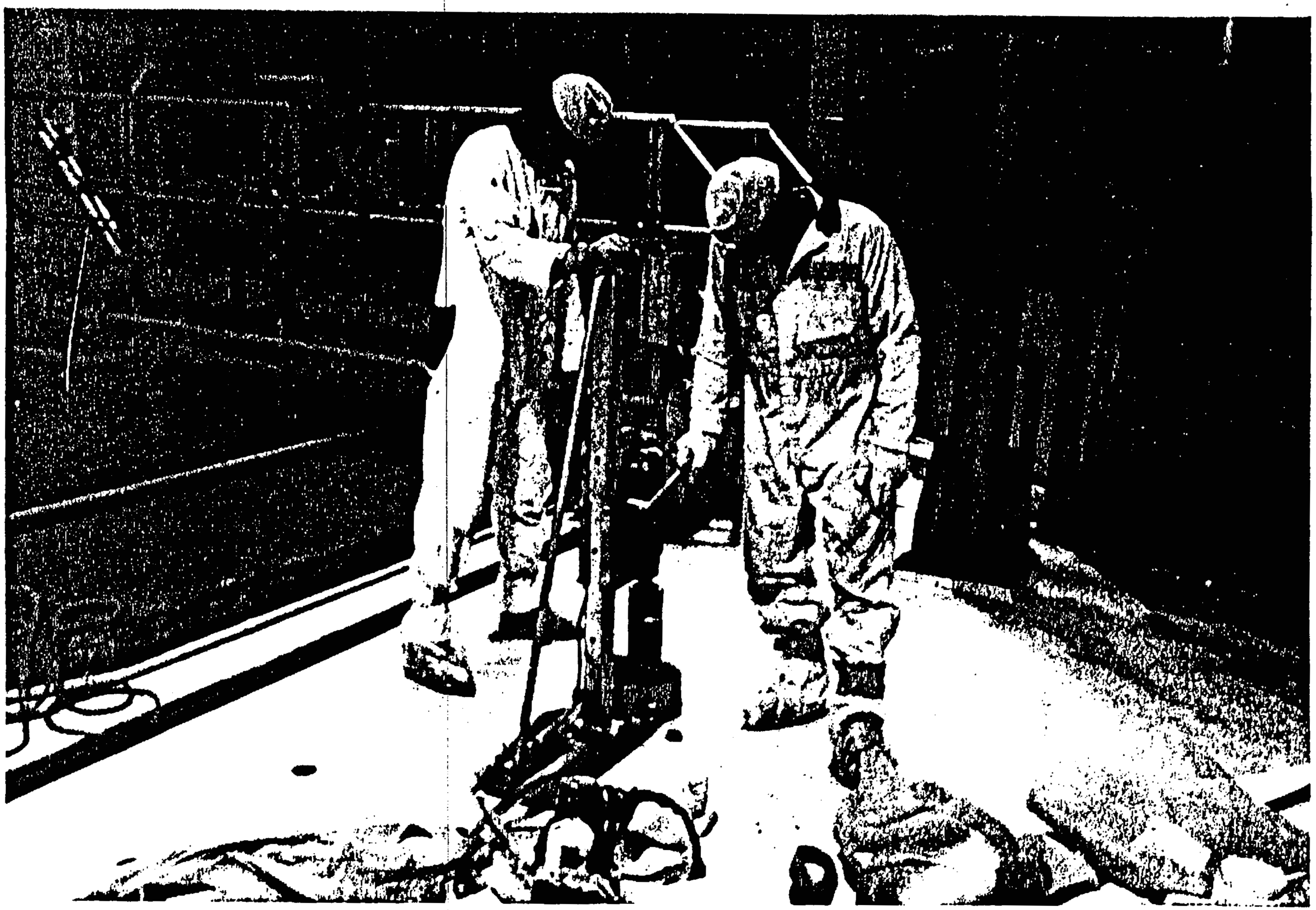


FIGURE 2.23. Concrete Core Drilling on Operation Floor of Reactor Building

FIGURE 2.24. Removing Concrete Core from Operating Floor in Reactor Building

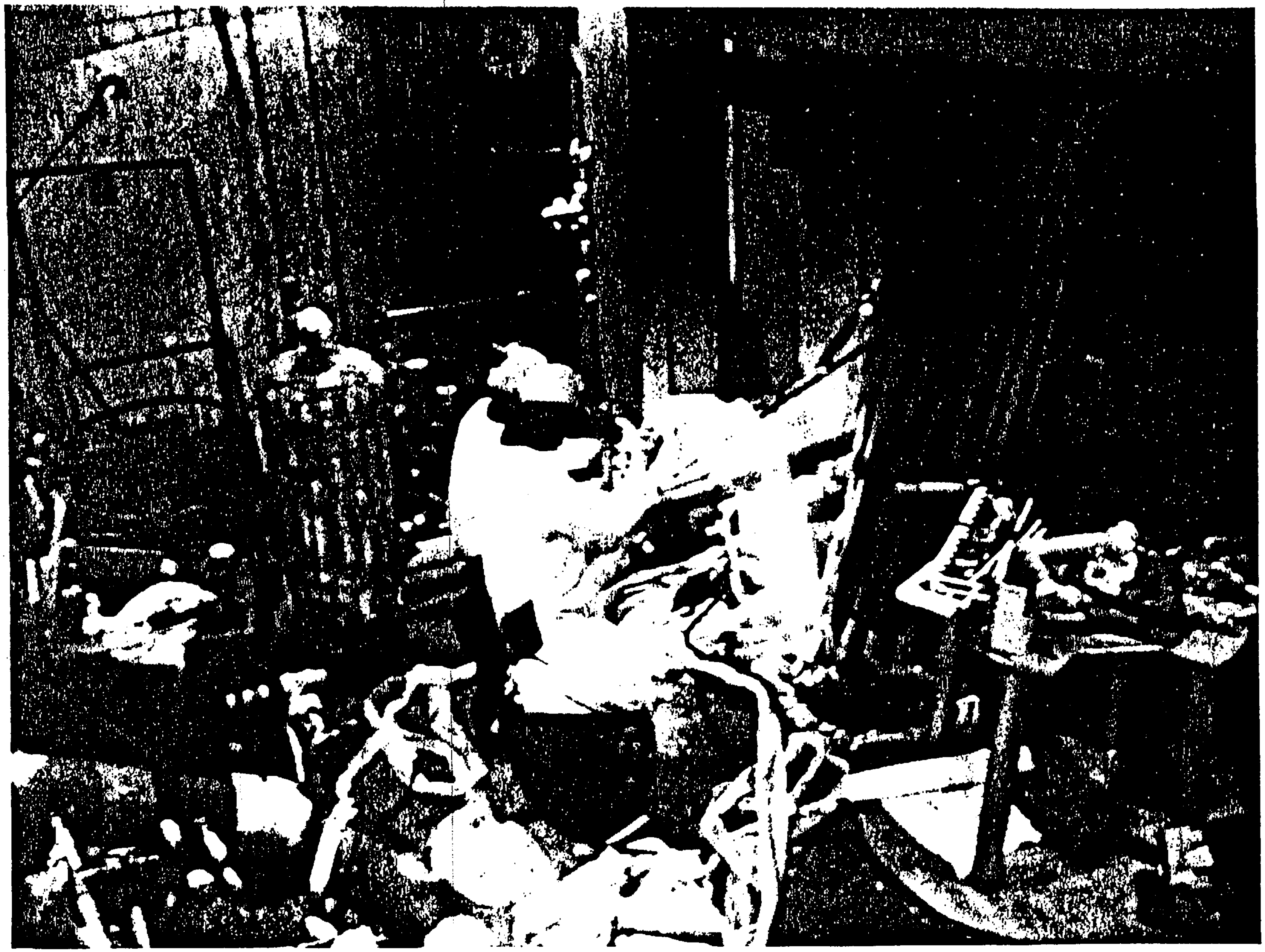


FIGURE 2.25. Coring Concrete Bioshield at Pathfinder

sufficient integrity that long sections of core (totaling about 2 m) could be removed intact. However, a void space exists in the bioshield wall about 2 m from the outside surface. This gap prevented the return flow of the cooling water to the recirculation reservoir, thus necessitating the use of large volumes of water if drilling was to proceed past this point. Since NSP had requested that we generate little or no liquid wastes during this program, we were forced to terminate the drilling before penetrating the entire bioshield and only about 2 m of core were collected.

2.5 SAMPLE HANDLING AND ANALYSIS

Upon return of the samples to our Richland laboratory, the 4- and 5-cm-dia. hole-saw pipe cores were repackaged in sealed plastic bags and counted directly on Ge(Li) gamma-ray spectrometers. The concrete cores were likewise repackaged and directly counted. The concrete cores were later cut into 1-cm slabs with a diamond blade lapidary saw to determine radionuclide contamination as a function of depth in the concrete.

The outside, uncontaminated surfaces of the hole-saw cores and the piping (cut into 5- to 10-cm lengths) were then coated with PDS® plastic coating prior to stripping off the corrosion film on the inside surfaces by immersion in hot 6 N HCl. The plastic coating minimized the amount of stable iron and other metal constituents leached from the samples. Large amounts of stable iron and nickel limit the sensitivity for measuring the low energy X-ray and beta emitters ^{55}Fe, ^{59}Ni, and ^{63}Ni, and large amounts of other metals can interfer with the sequential radiochemical separations used to measure other beta and alpha-emitting radionuclide. The contaminated hardware collected from the shield pool and the fuel storage pool contained absorbed radioactivity on all surfaces and was leached as described above without coating with plastic paint. The HCl leachates were evaporated on a hot plate to a volume of 100 ml, filtered of any insoluble matter and appropriate aliquots removed and counted directly on Ge(Li) gamma-ray spectrometers to measure the gamma-emitting radionuclides present in the samples. Radionuclides included in the direct Ge(Li) gamma-ray spectrometric computer program were ^{22}Na, ^{54}Mn, ^{60}Co, ^{65}Zn, ^{94}Nb, ^{106}Ru, ^{108m}Ag, ^{110m}Ag, ^{125}Sb, ^{126}Sn-Sb, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{152}Eu, ^{154}Eu, ^{155}Eu, ^{166}Ho and ^{228}Ra.

The filters were also counted to see if any radioactivity was associated with the insoluble matter which was sometimes present. Usually the filter contained only a few percent of the radioactivity compared to the leachate, except for ^{125}Sb which was consistently present on the filters of many of the samples.

Following the direct gamma-ray spectrometry, aliquots of leachate were taken for radiochemical analyses of ^{14}C, ^{55}Fe, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{94}Nb, ^{99}Tc, ^{238}Pu, ^{239}Pu, and ^{241}Am.

3.0 RADIONUCLIDE MEASUREMENTS

Radionuclide concentrations in soil, auxiliary structures, and in the nuclear plant piping and hardware are reported in this section.

3.1 SOIL

The radionuclide concentrations in soil samples collected around the Pathfinder plant are presented in Table 3.1. Of the ten samples collected, only four showed any indication of reactor originated radionuclides--Samples S-1, S-2, S-6 and S-10. Samples S-1 and S-2 were collected within a few feet of the reactor building (see Figure 2.4). Sample S-6 was a grab sample collected from the top 5 cm of the dried and cracked sediments contained in the water treatment effluent basin. Sample S-10 was a composite of four surface (top 2.5 cm) grab samples collected just behind the shipping dock of the fuel handling building. These four samples contained ^{60}Co concentrations of reactor origin, being somewhat elevated over the fallout background. However, the absolute concentrations of ^{60}Co were extremely low and none of the samples exceeded 2 pCi/g.

In samples S-1 and S-2, the ^{60}Co concentrations were elevated over the fallout background in only the top 2.5 cm of soil, and the radioactivity appeared to be heterogeneously distributed as small discrete particles. The grab sample from the water treatment effluent basin (S-6) contained the highest ^{60}Co concentration (1.7 pCi/g). This sample was composed of very fine textured sediments and mineral by-products of the water treatment backwash effluents. Since other long-lived fallout radionuclides (i.e., ^{137}Cs) were very low in this sediment, it can probably be concluded that the traces of ^{60}Co were of reactor origin. Soil sample S-10, which was collected near the shipping dock of the fuel handling building, contained ^{60}Co concentrations slightly higher than the fallout background.

The soil sample with the highest ^{137}Cs and $^{239-240}Pu$ content (S-4, top 2.5 cm) does not appear to have any detectable reactor origin radionuclides present. The ^{137}Cs and $^{239-240}Pu$ appear to be solely due to fallout from nuclear weapons testing. This conclusion is based on the isotopic ratio of

TABLE 3.1. Radionuclide Concentrations in Soil, Cooling Tower and Boiler Sludge from Pathfinder Generating Plant, July 1980

pCi/gm

Sample	Type	Depth (cm)	^{7}Be	^{40}K	^{51}Cr	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{95}Zr	^{95}Nb	^{103}Ru	^{106}Ru	^{125}Sb	^{134}Cs
S-1	Soil	0-2.5	<0.16	10.5	<0.74	0.0051	<0.013	0.521	<0.016	<0.026	<0.052	<0.034	0.10	<0.020	<0.0037
S-2	Soil	0-2.5	<0.19	7.90	<0.87	0.0069	<0.017	1.288	<0.020	0.098	<0.063	<0.042	<0.050	<0.024	<0.0045
S-2	Soil	2.5-7.6	0.33	8.38	<1.2	<0.0051	<0.014	0.021	<0.016	0.028	<0.054	<0.039	0.078	<0.022	<0.0040
S-3	Soil	0-2.5	<0.24	12.3	<1.1	0.0059	<0.016	0.024	<0.020	0.033	<0.069	<0.050	<0.054	<0.027	<0.0051
S-3	Soil	2.5-7.6	<0.26	12.4	<1.2	<0.0067	<0.018	0.022	<0.019	0.10	<0.073	<0.051	<0.057	<0.028	<0.0052
S-4	Soil	0-2.5	<0.41	15.4	<1.7	<0.0082	<0.022	0.045	<0.026	<0.044	<0.093	<0.082	<0.082	<0.042	<0.0077
S-4	Soil	2.5-7.6	<0.31	16.5	<1.4	0.012	<0.020	0.013	<0.024	0.13	<0.091	<0.065	<0.068	<0.035	<0.0066
S-5	Soil sample from effluent pitch	0-15	1.14	8.45	<2.3	<0.013	<0.033	<0.011	<0.043	<0.067	<0.15	<0.097	<0.10	<0.058	<0.011
S-6	Soil sample from water treatment eff. backwash basin	0-5	1.00	15.6	<6.6	0.063	<0.076	1.734	<0.098	0.21	<0.36	<0.20	0.36	<0.13	<0.023
S-7	Soil	0-2.5	<0.22	9.63	<1.0	<0.0065	<0.016	0.027	<0.020	0.082	<0.064	<0.045	<0.057	<0.029	<0.0053
S-7	Soil	2.5-7.6	<0.20	13.9	<0.87	<0.0057	<0.014	0.044	<0.019	0.046	<0.059	<0.042	<0.10	<0.026	<0.0050
S-8	Soil from fenced-in storage area	0-2.5	<0.16	7.95	<0.70	<0.0048	<0.012	0.019	<0.016	0.043	<0.048	<0.050	<0.040	<0.021	<0.0038
S-9	Soil	0-2.5	0.43	10.8	<1.1	<0.0062	<0.016	<0.007	<0.020	<0.033	<0.064	<0.051	0.086	<0.029	<0.0050
S-9	Soil	2.5-7.6	<0.25	15.1	<1.2	<0.0073	<0.020	<0.009	<0.024	0.074	<0.078	<0.054	<0.061	<0.032	<0.0059
S-10	Soil	0-2.5	<0.24	3.43	<1.1	<0.0070	<0.012	0.208	<0.014	<0.024	<0.049	<0.050	0.15	<0.034	<0.0049
CT-1	Sludge from cooling tower		<0.79	19.9	<3.6	<0.021	<0.055	<0.021	<0.080	<0.11	<0.26	<0.16	0.48	<0.10	<0.020
CT-2	Iron oxides from cooling tower		<0.18	1.30	<1.1	<0.0050	<0.014	0.016	<0.016	<0.027	<0.061	<0.038	<0.044	<0.024	<0.0044
CT-3	Wood slats from cooling tower		<1.5	25.62	<9.4	<0.039	<0.11	0.192	<0.12	0.58	<0.43	<0.31	<0.35	<0.18	<0.034
FBS-1	Fossil side-boiler sludge		<3.7	5.00	<15.0	0.21	<0.45	275.0	<1.5	1.50	<1.6	<0.85	<1.1	<0.51	<0.095
FBS-2	Fossil side-boiler sludge		<3.6	4.34	<19.0	<0.16	<0.44	334.0	<3.4	<0.74	<1.6	<0.85	<1.0	<0.50	<0.091
FBS-3	Fossil side-boiler sludge		<3.9	4.15	<17.0	<0.17	<0.47	256.0	<1.9	<0.81	<1.7	0.91	<1.2	<0.52	<0.096

TABLE 3.1. (contd)

Sample	Type	Depth (cm)	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{228}Th	^{226}Ra	^{228}Ra	^{238}Pu	$^{239-240}Pu$	^{241}Am
S-1	Soil	0-2.5	0.253	<0.071	0.035	<0.035	<0.026	<0.043	0.303	0.261	0.355	0.00041	0.00416	0.00159
S-2	Soil	0-2.5	0.224	0.14	0.044	<0.041	0.055	<0.045	0.397	0.219	0.376	0.00026	0.00290	0.00223
S-2	Soil	2.5-7.6	0.153	<0.089	<0.032	<0.040	<0.027	<0.017	0.381	0.202	0.341			
S-3	Soil	0-2.5	0.905	0.180	0.073	<0.051	<0.031	<0.077	0.486	0.495	0.494			
S-3	Soil	2.5-7.6	0.770	0.33	0.078	<0.055	<0.033	<0.075	0.811	0.494	0.622			
S-4	Soil	0-2.5	4.38	0.21	0.18	<0.082	<0.042	<0.111	0.680	0.653	0.743	0.00408	0.0698	0.00698
S-4	Soil	2.5-7.6	1.28	0.14	<0.044	<0.064	<0.040	<0.091	0.838	0.651	0.847			
S-5	Soil sample from effluent pitch	0-15	0.016	0.45	<0.069	<0.099	<0.062	<0.058	0.892	2.07	0.395			
S-6	Soil sample from water treatment eff. backwash basin	0-5	0.439	1.51	<0.20	<0.29	<0.14	<0.368	2.57	9.68	3.79	0.000027	0.00059	0.00022
S-7	Soil	0-2.5	0.455	0.12	0.079	<0.054	<0.033	<0.020	0.905	0.635	0.788			
S-7	Soil	2.5-7.6	0.720	<0.085	<0.033	<0.049	<0.030	<0.017	0.649	0.477	0.667			
S-8	Soil from fenced-in storage area	0-2.5	0.095	<0.076	<0.030	<0.038	<0.028	<0.018	0.441	0.293	0.439			
S-9	Soil	0-2.5	0.667	<0.11	<0.041	<0.054	<0.034	<0.082	0.797	0.495	0.689			
S-9	Soil	2.5-7.6	0.269	<0.11	<0.042	<0.059	<0.041	<0.070	0.730	0.494	0.721			
S-10	Soil	0-2.5	2.89	0.10	<0.10	<0.049	0.045	<0.016	0.327	0.293	0.327			
CT-1	Sludge from cooling tower		0.662	1.24	0.450	<0.18	<0.11	<0.075	1.32	4.43	2.00			
CT-2	Iron oxides from cooling tower		0.027	0.28	<0.034	<0.044	0.059	<0.016	0.256	0.743	0.309			
CT-3	Wood slats from cooling tower		0.274	1.29	<0.23	<0.35	0.421	<0.183	0.357	1.48	0.937			
FBS-1	Fossil side-boiler sludge		<0.10	0.98	<0.32	0.86	<1.0	<0.329	0.532	0.353	1.90			
FBS-2	Fossil side-boiler sludge		0.240	3.09	<0.41	1.95	<1.0	<0.14	0.626	0.191	1.00			
FBS-3	Fossil side-boiler sludge		0.545	3.2	0.68	1.66	<1.0	<0.15	0.272	0.477	1.42			

the ^{238}Pu to $^{239-240}Pu$ observed in this soil sample (0.0585), which is precisely the 1978-1979 average ratio of accumulated fallout plutonium in surface (top cm) soils at background locations in the U. S.[7] The residual plutonium concentrations in the Pathfinder plant piping and hardware were extremely low, and normally exhibited a $^{238}Pu/^{239-240}Pu$ ratio of approximately one. By examining the plutonium isotopic ratios in surface soils, it would be possible to detect as little as 2% to 5% contribution of reactor-origin plutonium in the presence of the more prevalent fallout plutonium because of their greatly different isotopic abundances. Thus, no evidence of ^{137}Cs and $^{239-240}Pu$ from the Pathfinder plant could be observed in the soils around the site. The ^{137}Cs and $^{239-240}Pu$ concentrations in soil sample S-4 were typical of the background levels observed in relatively undisturbed soils containing only fallout radionuclides.[7]

The soils around the plant also occasionally contained traces of ^{54}Mn, ^{95}Zr, ^{106}Ru, ^{141}Ce and ^{144}Ce. However, these radionuclides are undoubtedly of fallout origin and are present in trace concentrations typically observed in background locations.

The ^{40}K, ^{228}Th, ^{226}Ra and ^{228}Ra concentrations in the soils were typical of naturally-occurring levels, except for sample S-6. The unusually high concentration of Th and Ra is probably due to impurities in the chemicals used in the water treatment process, i.e., alum.

It was surprising that no reactor-origin radionuclides, particularly ^{60}Co, were observed in the drainage ditch which had received low-level radwaste discharges during reactor operations. Apparently, the ditch has cleansed itself of any traces of radionuclides by the continuous flushing with uncontaminated water that has occurred since the conversion to the fossil-fuel unit.

3.2 COOLING TOWER

Samples of wood slats, rust deposits and sludge from the bottom of the cooling tower were analyzed for the spectrum of long-lived radionuclides (see Table 3.2). Only the wood slats contained a trace of ^{60}Co (0.19 pCi/g). However, this could be due to accumulation of fallout radioactivity since

TABLE 3.2. Residual Radionuclide Concentrations on Piping and Hardware Surfaces at Pathfinder, September 1980

pCi/cm² - (pCi/gm) in parentheses

Sample	Identification	^{22}Na	^{54}Mn	^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
MSL-A	Main steam line 16" carbon steel	<0.36 (<0.026)	<0.39 (<0.027)	523 (37.7)	4.28 (0.308)	1369 (98.6)	1080 (77.5)	<12 (<0.86)	<0.29 (<0.021)	<2.6 (<0.19)	3.43 (0.247)	<1.5 (<0.104)	<0.61 (<0.044)
MSB-B	Main steam bypass line	*	*	994 (205)	8.13 (1.68)	5666 1170	2051 424	*	<0.18 (<0.037)	*	*	*	*
WP-74-301A	Reactor feeowater line 8" dia. carbon steel	*	*	615 (96.4)	14.1 (2.20)	19350 (3016)	6346 (989)	*	<0.51 (<0.079)	*	*	*	*
RFW-A	Reactor feeowater line 8" dia. carbon steel	*	*	1465 (150)	15.0 (1.54)	21219 (2177)	4618 (474)	*	<0.170 (<0.017)	*	*	*	*
WP-42 11	Reactor feedwater line 8" diameter carbon steel	<0.092 (<0.014)	<0.0820 (<0.013)	1465 (228)	15.0 (2.34)	21219 (3308)	4618 (720)	<0.250 (<0.039)	<0.067 (<0.0103)	<0.59 (<0.093	<0.096 (<0.015)	<0.40 (<0.062)	<0.16 (<0.026)
WP-76-300-A3	Reactor water purification line	<1.5 (<0.34)	<1.5 (<0.35)	923 (210)	2.32 (0.531)	10117 (2309)	722 (165)	<78 (<18)	<1.2 (<0.27)	10 (<2.4)	<0.83 (<0.19)	<5.8 (<1.3)	<2.5 (<0.57)
IFDP	Inlet line to reactor water demin. purification system	<0.42 (<0.090)	<0.70 (<0.15)	*	*	1806 (387)	*	<5.9 (<1.3)	<0.372 (<0.080)	<3.3 (<0.71)	<0.26 (<0.056)	<1.8 (<0.39)	<0.79 (<0.17)
RSA	Reactor sump pump line 1 7/8" diameter stainless steel	<0.020 (<0.0090)	<0.021 (<0.0095)	*	*	33.2 15.1	*	<0.31 (<0.14)	<0.017 (<0.0077)	<0.16 (<0.071)	5.18 (2.34)	<0.079 (<0.036)	<0.042 (<0.019)
RSE	Reactor sump pump line carbon steel elbow	<0.44 (<0.043)	<0.47 (<0.045)	235 (23)	1.32 (0.13)	1669 (161)	445 (42.8)	<16 (<1.5)	<0.017 (<0.0016)	<3.2 (<0.31)	69.8 (6.71)	<1.8 (<0.17)	<0.81 (<0.077)
RSE-Pipe	Reactor sump pump line 1 1/2" dia. SS pipe from RSE	<0.076 (<0.025)	<0.14 (<0.48)	77.9 (26.0)	0.223 (0.0743)	120 (40.0)	61.3 (20.4)	<1.0 (<0.34)	<0.023 (<0.0077)	<0.50 (<0.16)	108 (40.0)	<0.21 (<0.070)	<0.19 (<0.062)
RLL-A	Reactor liquid level column 2 3/8" dia. stainless steel	<0.75 (<0.15)	<0.83 (<0.17)	14996 (2999)	11.8 (2.36)	4020 (804)	4163 (832)	<31 (<6.2)	<0.014 (<0.0027)	<5.6 (<1.1)	<0.45 (<0.091)	<3.2 (<0.63)	<1.4 (<0.27)
SPC-A	Shield pool cleanup line 4" dia. carbon steel	<0.203 (<0.082)	<0.23 (<0.093)	2803 (1135)	1.47 (0.595)	782 (317)	480 (195)	<5.1 (<2.09)	<0.011 (<0.0045)	<1.6 (<0.64)	<0.13 (<0.051)	<1.3 (<0.53)	<0.38 (<0.16)
PWC-B	Pool water cleanup line 2" dia. stainless steel (storage shield pools)	<0.68 (<0.27)	<0.73 (<0.29)	2544 (1006)	5.36 (2.13)	23601 (9335)	1646 (651)	<21 (<8.2)	<0.031 (<0.012)	<4.95 (<1.95)	<0.38 (<0.15)	<2.8 (<1.1)	<1.2 (<0.46)

* Not measured.

TABLE 3.2. (contd)

Radionuclide Concentrations - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ra	^{238}Pu	^{239}Pu	^{241}Am
MSL-A	Main steam line 16" carbon steel	<0.23 (<0.017)	<0.18 (<0.013)	<0.43 (<0.031)	<0.84 (<0.060)	2.52 0.181	<4.8 (<0.34)	<0.76 (<0.055)	<0.54 (<0.039)	<4.3 (<0.31)	0.00143 (0.000104)	0.00155 (0.00011)	*
MSB-B	Main steam bypass line	*	*	0.0822 (0.0169)	*	*	*	*	*	*	0.00272 (0.000562)	0.00294 (0.000608)	*
WP-74-301A	Reactor feeowater line 8" dia. carbon steel	*	*	0.0905 0.0141	*	*	*	*	*	*	0.00119 (0.000185)	0.00215 (0.000338)	*
RFW-A	Reactor feedwater line 8" dia. carbon steel	*	*	<0.039 (<0.0040)	*	*	*	*	*	*	0.000964 (0.0000988)	0.00778 (0.000183)	*
WP-42 11	Reactor feedwater line 8" diameter carbon steel	<0.057 (<0.090)	<0.045 (<0.0072)	<0.074 (<0.0012)	<0.42 (<0.066)	<0.28 (<0.043)	<0.97 (<0.15)	<0.15 (<0.024)	<9.6 (<1.5)	<0.91 (<0.14)	0.000964 (0.000149)	0.00178 (0.000279)	*
WP-76-300-A3	Reactor water purification line	<0.87 (<0.20)	<0.72 (<0.16)	3.48 (0.793)	<3.9 (<0.88)	<4.2 (<0.97)	<19 (<4.3)	<1.8 (<0.42)	<2.2 (<0.50)	<17 (<3.9)	0.000279 (0.000063)	0.000324 (0.000072)	0.00775 (0.00177)
IFDP	Inlet line to reactor water demin. purification system	<0.27 (<0.059)	<0.23 (<0.049)	0.914 (0.196)	<3.4 (<0.73)	<1.1 (<0.23)	<5.9 (<1.3)	<0.58 (<0.12)	<0.69 (<0.15)	<5.4 (<1.1)	*	*	*
RSA	Reactor sump pump line 1 7/8" diameter stainless steel	<0.019 (<0.0086)	<0.011 (<0.0050)	<0.019 (<0.0086)	<0.058 (<0.026)	<0.048 (<0.022)	<0.26 (<0.12)	<0.027 (<0.012)	<0.031 (<0.014)	<0.24 (<0.11)	*	*	*
RSE	Reactor sump pump line carbon steel elbow	<0.28 (<0.027)	<0.22 (<0.021)	0.820 (0.079)	<1.6 (<0.16)	4.59 (0.44)	<5.9 (<0.56)	<0.51 (<0.049)	<0.67 (<0.064)	<5.3 (<0.51)	0.000392 (0.000036)	0.00248 (0.000239)	0.00117 (0.000113)
RSE-Pipe	Reactor sump pump line 1 1/2" dia. SS pipe from RSE	<0.062 (<0.021)	<0.036 (<0.012)	<0.056 (<0.018)	<0.58 (<0.19)	0.85 (0.28)	<0.68 (<0.22)	<0.12 (<0.041)	<0.093 (<0.031)	<0.61 (<0.204)	0.00108 (0.000365)	0.00676 (0.00221)	*
RLL-A	Reactor liquid level column 2 3/8" dia. stainless steel	<0.47 (<0.094)	<0.39 (<0.078)	2.02 (0.405)	<3.7 (<0.73)	<2.02 (<0.404)	<10.2 (<2.03)	<0.99 (<0.20)	<1.2 (<0.24)	<9.2 (<1.9)	0.000450 (0.000090)	0.00117 (0.000225)	*
SPC-A	Shield pool cleanup line 4" dia. carbon steel	<0.13 (<0.053)	<0.11 (<0.044)	1.35 (0.545)	<0.79 (<0.32)	<0.52 (<0.21)	<2.8 (<1.1)	<0.27 (<0.11)	<0.33 (<0.13)	<2.5 (<1.0)	0.00901 (0.00369	0.0563 (0.0230	0.0649 (0.0261)
PWC-B	Pool water cleanup line 2" dia. stainless steel (storage & shield pools)	<0.41 (<0.16)	<0.34 (<0.13)	40.2 (15.9)	<1.6 (<0.62)	<1.7 (<0.66)	<9.1 (<3.6)	<0.97 (<0.39)	<1.03 (<0.41)	<8.3 (<3.3)	0.0315 (0.0126)	0.212 (0.0838)	0.104 (0.0410)

* Not measured.

TABLE 3.2. (contd)

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{22}Na	^{54}Mn	^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
PDO-A	Outlet from pool demin. 2" dia. SS pipe	<0.22 (<0.085)	<0.11 (<0.041)	*	*	535 (204)	*	<4.8 (<1.8)	<0.081 (<0.031)	<0.72 (<0.27)	<0.055 (<0.,021)	<0.40 (<0.15)	<0.35 (<0.14)
PDI-A	Inlet to pool demin. 2" dia. SS pipe	<0.25 (<0.104)	<0.48 (<0.196)	*	*	1346 (555)	*	<12 (<4.95)	<0.36 (<0.15)	<3.2 (<1.3)	<0.25 (<0.101)	<1.8 (<0.74)	<0.797 (<0.33)
SPCB-A	Shield pool coolant bypass 4" dia. carbon steel pipe	<0.32 (<0.069)	<0.35 (<0.075)	*	*	873 (191)	*	<8.7 (<1.9)	<0.26 (<0.057)	<2.33 (<0.509)	<0.18 (<0.0401)	<1.3 (<0.29)	<0.55 (<0.12)
SPD-A	Shield pool drain pipe 4" dia. SS pipe	<3.7 (<1.6)	<3.96 (<1.7)	338 (142)	0.572 (0.2405)	4599 (1931)	173 (72.5)	<34 (<14)	<0.59 (<0.025)	<45 (<19)	<2.3 (<0.95)	<15 (<6.4)	<6.3 (<2.6)
TR-B	Tool rack from shield pool-strip of SS plate	<0.0405 (<0.0211)	<0.046 (<0.025)	*	*	123 (64)	*	<1.19 (<0.63)	<0.0405 (<0.021)	<0.33 (<0.17)	0.14 (0.075)	<0.18 (<0.097)	<0.081 (<0.043)
TR-B-NW	Nuts and washers from TR-B-stainless steel	<2.1 (<0.82)	<2.3 (<0.901)	*	*	883 (332)	*	<7.9 (<3.1)	<3.7 (<1.4)	16 (<6.4)	<1.3 (<0.51)	<9.1 (<3.5)	<8.9 (<3.3)
FHR-A	Fuel shoot support strut from shield pool - 2 1/2" dia.	<0.86 (<0.21)	<0.99 (<0.24)	55.0 (13.4)	<0.100 (<0.024)	788 (192)	35.3 (8.603)	<4.8 (<1.2)	<0.019 (<0.0050)	<6.5 (<1.6)	<0.63 (<0.15)	<5.7 (<1.4)	<1.5 (<0.37)
SHFBS-A	Superheater fuel storage rack - 1" dia. SS tubes	<0.17 (0.36)	<0.19 (<0.41)	*	*	101 (219)	*	<0.59 (<1.27)	<0.14 (<0.31)	<1.3 (<2.9)	<0.103 (<0.22)	<0.72 (<1.5)	<0.32 (<0.69)
SHFBS-C	Same as SHFBS-A, except from another bundle	<0.53 (<1.1)	<0.91 (<1.97)	*	*	178 (386)	*	<2.7 (<5.9)	<0.45 (<0.98)	4.1 (<8.9)	<0.32 (<0.69)	<2.5 (<5.3)	<1.2 (<2.5)
FTC	Fuel transfer chute in fuel storage basin - SS plate	<0.16 (<0.089)	<0.302 (<0.17)	67.6 (37.4)	<0.19 (<0.105)	963 (534)	22.9 (12.7)	<5.7 (<3.1)	<0.071 (<0.039)	<1.20 (<0.67)	<0.17 (<0.094)	<0.68 (<0.37)	<0.43 (<0.24)
FTTR	Fuel transfer chute roller wheel - SS roller	<0.59 (<0.14)	<0.67 (<0.15)	*	*	547 (127)	*	<2.1 (<0.500)	<0.504 (<0.12)	<4.5 (<1.04)	0.91 (0.21)	<2.5 (<0.57)	<1.04 (<0.24)
FRSB-A	Fuel storage rack - 2" dia. SS cylinder - east end	<3.1 (<1.5)	<2.4 (<1.2)	294 (145)	0.356 (0.176)	4274 (2113)	117 (57.7)	<32 (<16)	<0.018 (<0.0090)	<16 (<7.97)	<1.3 (<0.65)	<9.01 (<4.5)	<4.04 (<1.99)
FRSB-C	Same as FRSB-A, except from west end	<3.9 (<2.02)	<3.98 (<2.04)	537 (275)	0.599 (0.308)	8252 (4234)	188 (96.5)	<48 (<25)	<0.025 (<0.013)	<27 (<14)	<2.2 (<1.1)	<15 (<7.8)	<6.6 (<3.4)

* Not measured.

TABLE 3.2. (contd)

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ra	^{238}Pu	^{239}Pu	^{241}Am
PDO-A	Outlet from pool demin. 2" dia. SS pipe	<0.059 (<0.023)	<0.049 (<0.019)	0.245 (<0.0932)	<0.402 (<0.15)	<0.23 (<0.086)	<1.3 (<0.51)	<0.105 (<0.0401)	<0.15 (<0.058)	<1.2 (<0.058)	*	*	*
PDI-A	Inlet to pool demin. 2" dia. SS pipe	<0.26 (<0.108)	<0.23 (<0.094)	1.27 (0.523)	<2.3 (<0.94)	<1.05 (<0.44)	<5.9 (<2.4)	<0.47 (<0.19)	<0.77 (<0.32)	<5.4 (<2.2)	*	*	*
SPCB-A	Shield pool coolant bypass 4" dia. carbon steel pipe	<0.19 (<0.042)	<0.16 (<0.035)	<0.29 (<0.064)	<0.75 (<0.16)	<0.75 (<0.16)	<4.3 (<0.94)	<0.38 (<0.083)	<0.49 (<0.11)	<3.9 (<0.85)	*	*	*
SPD-A	Shield pool drain pipe 4" dia. SS pipe	<2.2 (<0.92)	<1.8 (<0.76)	<8.6 (<3.6)	<8.4 (<3.5)	<9.1 (<3.8)	<55 (<23)	<3.9 (<1.6)	<5.6 (<2.4)	<45 (<19)	<0.000900 (<0.0000360)	<0.0000631 (<0.0000270)	
TR-B	Tool rack from shield pool- strip of SS plate	<0.028 (<0.015)	<0.024 (<0.013)	<0.0405 (<0.021)	<0.13 (<0.067)	<0.24 (<0.13)	<1.2 (<0.63)	<0.059 (<0.031)	<0.068 (<0.036)	<0.53 (<0.28)	*	*	*
TR-B-NW	Nuts and washers from TR-B- stainless steel	<2.6 (<0.995)	<1.3 (<0.49)	<2.0 (<0.78)	<6.4 (<2.4)	<4.8 (<1.9)	<31 (<12)	<3.0 (<1.1)	<3.5 <1.3	<27 (<10.4)	*	*	*
FHR-A	Fuel shoot support strut from shield pool - 2 1/2" dia	<0.55 (<0.13)	<0.45 (<0.11)	<0.83 (<0.201)	<2.1 (<0.504)	<2.1 (<0.51)	<12 (<2.9)	<1.6 (<0.39)	<1.4 (<0.34)	<11 (<2.7)	0.000315 (0.000901)	0.000766 (0.000180)	0.00302 (0.000721)
SHFBS-A	Superheater fuel storage rack - 1" dia. SS tubes	<0.11 (<0.24)	<0.091 (<0.20)	1.23 (2.65)	<0.48 (<1.04)	<0.43 (<0.92)	<2.3 (<5.00)	<0.24 (<0.51)	<0.38 (<0.82)	<2.1 (<4.5)	*	*	*
SHFBS-C	Same as SHFBS-A, except from another bundle	<0.36 (<0.78)	<0.41 (<0.88)	<0.51 (<1.1)	<1.3 (<2.8)	<1.5 (<3.2)	<7.7 (<17)	<0.62 (<1.3)	<0.89 (<1.9)	<6.7 (<14.5)	0.000541 (0.00113)	<0.000045 (<0.000090)	0.00838 (0.01802)
FTC	Fuel transfer chute in fuel storage basin - SS plate	<0.23 (<0.13)	<0.085 (<0.047)	1.40 (0.775)	<0.39 (<0.21)	<0.38 (<0.21)	<2.2 (<1.2)	<0.203 (<0.11)	<0.41 (<0.23)	<1.98 (<1.10)	0.00131 (0.000721)	0.0000676 (0.0000360)	*
FTTR	Fuel transfer chute roller wheel - SS roller	<0.38 (<0.088)	<0.304 (<0.0703)	<0.58 (<0.13)	<1.7 (<0.399)	<1.4 (<0.33)	<8.2 (<1.9)	<0.68 (<0.16)	<0.93 (<0.21)	<7.4 (<1.7)	0.00189 (0.000450)	0.0000901 (0.0000225)	*
FRSB-A	Fuel storage rack - 2" dia. SS cylinder - east end	<1.3 (<0.66)	<1.1 (<0.56)	7.7 (3.8)	<5.9 (<2.96)	<5.99 (<2.95)	<29 (<14)	<2.8 (<1.4)	<3.4 (<1.7)	<25.6 (<12.7)	0.00946 (0.00464)	0.00149 (0.000766)	*
FRSB-C	Same as FRSB-A, except from west end	<2.9 (<1.5)	<1.9 (<0.97)	25.3 (13.00)	<9.99 (<5.1)	<12 (<5.98)	<49 (<25)	<4.7 (<2.4)	<5.7 (<2.9)	<44 (<23)	0.0296 (0.0152)	0.00401 (0.00206)	*

* Not measured.

TABLE 3.2. (contd)

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{22}Na	^{54}Mn	^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
FSBTT	Fuel transfer tube from fuel storage basin - 2" dia.	<0.71 (<0.16)	<0.78 (<0.18)	*	*	769 (178)	*	<3.7 (<0.87)	<0.59 (<0.14)	<5.3 (<1.2)	<0.42 (<0.097)	<4.2 (<0.97)	<1.3 (<0.29)
HSHT-Pipe	High solids holdup tank discharge line - 1 1/2" dia. SS pipe	<22 <(8.9)	<16 (<6.6)	886 (362)	0.34 (0.14)	13393 (5474)	119 (49)	<124 (<51)	<0.0300 (<0.012)	<108 (<44)	<8.9 (<3.6)	<60 (<25)	<26 (<11)
HSHT-Elbow	Carbon steel elbow from discharge line from HSHT	<1.3 (<0.55)	<1.5 (<0.64)	887 (389)	1.23 (0.5405)	9351 (4104)	402 (176)	<7.3 (<3.2)	<0.031 (<0.014)	<8.8 (<3.9)	<0.71 (<0.31)	<4.9 (<2.1)	<2.8 (<1.2)
CWTD-A	Concentrated waste tank discharge line - 1 1/2" dia. SS pipe	<0.029 (<0.013)	<0.041 (<0.018)	*	*	57.5 (24.9)	*	<0.401 (<0.17)	<0.025 (<0.011)	<0.23 (<0.098)	<0.018 (<0.0080)	<0.12 (<0.054)	<0.056 (<0.024)
SRTD	Spent resin tank discharge line - 2" dia. SS pipe	<0.403 (<0.12)	<0.63 (<0.19)	*	*	930 (275)	*	<8.7 (<2.6)	<0.33 (<0.099)	<2.99 (<0.88)	<0.46 (<0.14)	<2.5 (<0.75)	<0.703 (<0.21)
SCDL	Steam condensate drain line 1" dia. carbon steel line	<0.036 (<0.013)	<0.041 (<0.015)	*	*	67.6 (24.8)	*	<0.46 (<0.17)	<0.032 (<0.012)	<0.28 (<0.103)	0.0864 (0.0317)	<0.16 (<0.058)	<0.069 (<0.025)
Estimated conc. for Rad-waste system SS piping		<0.5	<0.5	300	2	5000	200	<5	<0.05	<10	<0.5	<2	<0.7
Estimated conc. for Rad-waste system carbon steel piping		<0.2	<0.2	1500	2	10000	400	<5	<0.05	<10	<0.5	<2	<0.7

* Not measured.

TABLE 3.2. (contd)

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ra	^{238}Pu	^{239}Pu	^{241}Am
FSBTT	Fuel transfer tube from fuel storage basin - 2" dia.	<0.45 (<0.104)	<0.36 (<0.083)	42.3 (9.82)	<2.9 (<0.67)	<1.7 (<0.39)	<16 (<3.6)	<0.804 (<0.19)	<1.1 (<0.25)	8.6 (<1.99)	*	*	*
HSHT-Pipe	High solids holdup tank discharge line - 1 1/2" dia. SS pipe	<9.01 (<3.7)	<9.2 (<3.8)	72.7 (29.7)	<40 (<16)	<51 (<21)	<234 (<95)	<32 (<13)	<23 (<9.2)	<60 (<25)	2.89 (1.18)	0.438 (0.179)	*
HSHT-Elbow	Carbon steel elbow from discharge line from HSHT	<0.74 (<0.32)	<0.61 (<0.27)	40.2 (17.6)	<3.3 (<1.4)	<3.7 (<1.6)	<16 (<6.9)	<1.5 (<0.68)	<1.8 (<0.81)	<14 (<6.3)	0.153 (0.0672)	1.02 (0.449)	0.641 (0.282)
CWTD-A	Concentrated waste tank discharge line - 1 1/2" dia. SS pipe	<0.019 (<0.0083)	<1.58 (<0.68)	0.236 (0.102)	<0.088 (<0.038)	<0.069 (<0.0300)	<0.401 (<0.17)	<0.042 (<0.018)	<0.047 (<0.0203)	<0.36 (<0.16)	*	*	*
SRTD	Spent resin tank discharge line - 2" dia. SS pipe	<0.25 (<0.073)	<0.203 (<0.0601)	<0.38 (<0.11)	<0.95 (<0.28)	<0.94 (<0.28)	<5.5 (<1.6)	<0.56 (<0.16)	<0.63 (<0.19)	<5.02 (<1.5)	*	*	*
SCDL	Steam condensate drain line 1" dia. carbon steel line	<0.024 (<0.0088)	<0.0197 (<0.0072)	<0.066 (<0.024)	<0.18 (<0.066)	<0.0897 (<0.033)	<0.495 (<0.18)	<0.059 (<0.022)	<0.059 (<0.021)	<0.45 (<0.17)	*	*	*
Estimated conc. for Rad-waste system SS piping		<0.3	<0.2	10	<1	<1	<1	<1	<1	<1	0.05	0.05	*
Estimated conc. for Rad-waste system carbon steel piping		<0.3	<0.2	10	<1	<1	<1	<1	<1	<1	0.05	0.05	*

* Not measured.

cooling towers can be rather efficient scrubbers of radioactive fallout. Despite the contamination of the cooling tower with reactor primary water in 1967, no significant residual radioactivity (either from the reactor or from fallout) presently remains in the cooling tower.

3.3 AUXILIARY BUILDINGS

The former construction administration building, the warehouse, and several small equipment sheds were surveyed with a G-M counter and smears were taken of the floors. Since no detectable radioactivity was observed during these surveys, no further sampling and analyses were conducted at these sites.

During our on-site work, July, 1980, it was possible to sample the sludge deposits in the fossil-fueled boilers presently used in the plant. This sludge contained 256 to 334 pCi/g of ^{60}Co (see Table 3.1). The origin of this ^{60}Co is presumed to be previously contaminated condensate and feedwater piping surfaces left intact and now used in the fossil-fired plant. Over a period of years, some of the ^{60}Co desorbed from these surfaces and became deposited in the boiler sludge.

3.4 NUCLEAR PLANT

Piping, hardware, and concrete within the nuclear steam supply system were examined for residual radionuclides. The measurements made on piping and hardware are discussed in Section 3.4.1. The measurements on concrete are discussed in Section 3.4.2.

3.4.1 Piping and Hardware

The residual radionuclide concentrations contained on various piping and hardware systems are presented in Table 3.2. Radionuclide concentrations are presented on a surface area basis as pCi/cm^2, and on a weight basis (numbers in parentheses) as pCi/g. The most abundant radionuclide present in the radioactive corrosion film on the piping and hardware was ^{60}Co (5.27 yr) followed by ^{63}Ni (100 yr), ^{55}Fe (2.7 yr), and ^{59}Ni (8×10^4 yr). At the time the Pathfinder reactor was shut down (1967), the major gamma-emitting radionuclide was ^{65}Zn (244 d). This radionuclide originated from zinc being corroded from the admiralty brass condenser and transported to the reactor via the feedwater, where the zinc became neutron-activated and redistributed in various piping systems. The ^{65}Zn has now all decayed to undetectable concentrations.

Because ^{60}Co was frequently the only significant gamma-emitting radionuclide present in the corrosion films, the gamma-ray spectra usually resembled a pure ^{60}Co spectrum (see Figure 3.1). Since there were apparently no fuel element failures during the 42 months of reactor operations, the corrosion films usually contained very little fission products. Cesium-137 was occasionally detected in very low concentrations in several piping systems and on hardware from the fuel storage pool. However, much of this ^{137}Cs was thought to have been brought into the plant in contaminated fuel shipping casks. Likewise, the ^{238}Pu, $^{239-240}Pu$, and ^{241}Am concentrations in contaminated surfaces were extremely low, and it is not certain how much of these transuranic radionuclides originated in Pathfinder by activation of tramp uranium on the fuel element surfaces or was brought into the plant in contaminated shipping casks.

The carbon steel reactor feedwater lines contained the highest radionuclide concentrations. The corrosion film deposited inside these pipes was substantially thicker than in the stainless steel piping, and the iron oxide surfaces in the feedwater lines were more efficient scavengers of radionuclides contained in the reactor feedwater. The main steam lines were also composed of carbon steel but contained much lower radionuclide concentrations compared to the feedwater lines. The steam lines appeared to have less corrosion product deposition, and therefore fewer adsorption sites for radionuclides. Also, the steam flow may have acted as a scouring mechanism to inhibit radionuclide deposition on the piping surfaces.

Another relatively contaminated piping loop was the pool water clean-up line (sample PWC-B). This piping (2-in.-dia stainless steel) not only contained relatively high residual concentrations of ^{55}Fe, ^{59}Ni, ^{60}Co, and ^{63}Ni, but also contained ^{137}Cs.

The reactor liquid level column piping (2-3/8-in. stainless steel) contained relatively high concentrations of ^{55}Fe, ^{59}Ni, and ^{63}Ni, although the ^{60}Co levels were not unusually high.

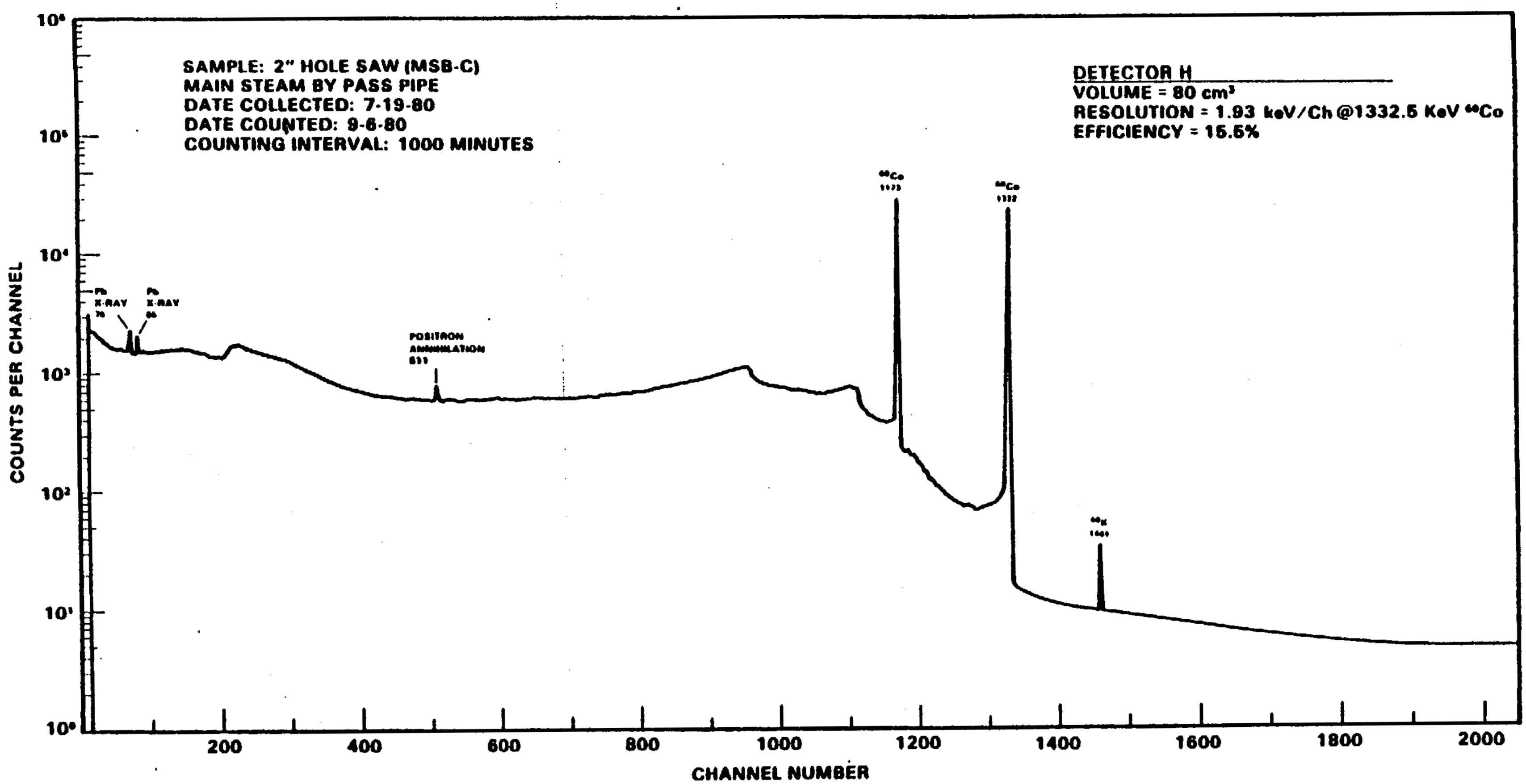


FIGURE 3.1. Gamma-Ray Spectrum of Hole-Saw Plug from Main Steam Bypass Line

Stainless steel piping from the reactor purification and rad-waste processing systems contained only intermediate levels of residual radioactivity, except for the high solids holdup tank. Piping coming from the high solids holdup tank contained a residue of loosely bound material which exhibited relatively high concentrations of ^{55}Fe, ^{59}Ni, ^{60}Co, and ^{63}Ni.

Samples of stainless steel hardware from the shield pool and fuel storage pool contained intermediate levels of adsorbed radioactivity. Samples from the fuel storage pool usually contained traces of adsorbed ^{137}Cs, as well as the usual spectrum of activation products.

The long-lived silver radionuclide, ^{108m}Ag (130 yr), was observed in several piping systems, including the main steam line and reactor sump piping. In sample RSE-pipe (1-1/2-in.-dia stainless steel pipe from the reactor sump line) the ^{108m}Ag was nearly as abundant as the ^{60}Co. It is curious that the ^{108m}Ag was observed only in these systems. To our knowledge, no silver or silver alloy components were used in the construction materials of the Pathfinder plant, and it appears that the ^{108m}Ag originated from neutron activation of trace amounts of silver impurities in the brass condenser in the stainless and carbon steels, or the zircaloy-clad boiler fuel elements.

In a previous study assessing the decommissioning of a PWR station,[1] it was postulated that the radionuclides that would be the principal contributors to external occupational radiation exposure (approximately 100 years after shutdown) would be ^{59}Ni (8×10^4 yr) and ^{94}Nb (2×10^4 yr). This does not appear to be entirely the case at Pathfinder. Figures 3.2, 3.3, and 3.4 present the change in concentration with time of residual radionuclides in three important piping systems at Pathfinder. Niobium-94 was never detected in any of the residual radioactivity translocated from the reactor pressure vessel, and would appear to be totally insignificant from radiation dose considerations. Nickel-63 concentrations were over two orders of magnitude higher than ^{59}Ni, but from external dose considerations are of lesser importance. In some piping systems, ^{137}Cs was abundant enough to provide the greatest external radiation dose after 100 years following shutdown.

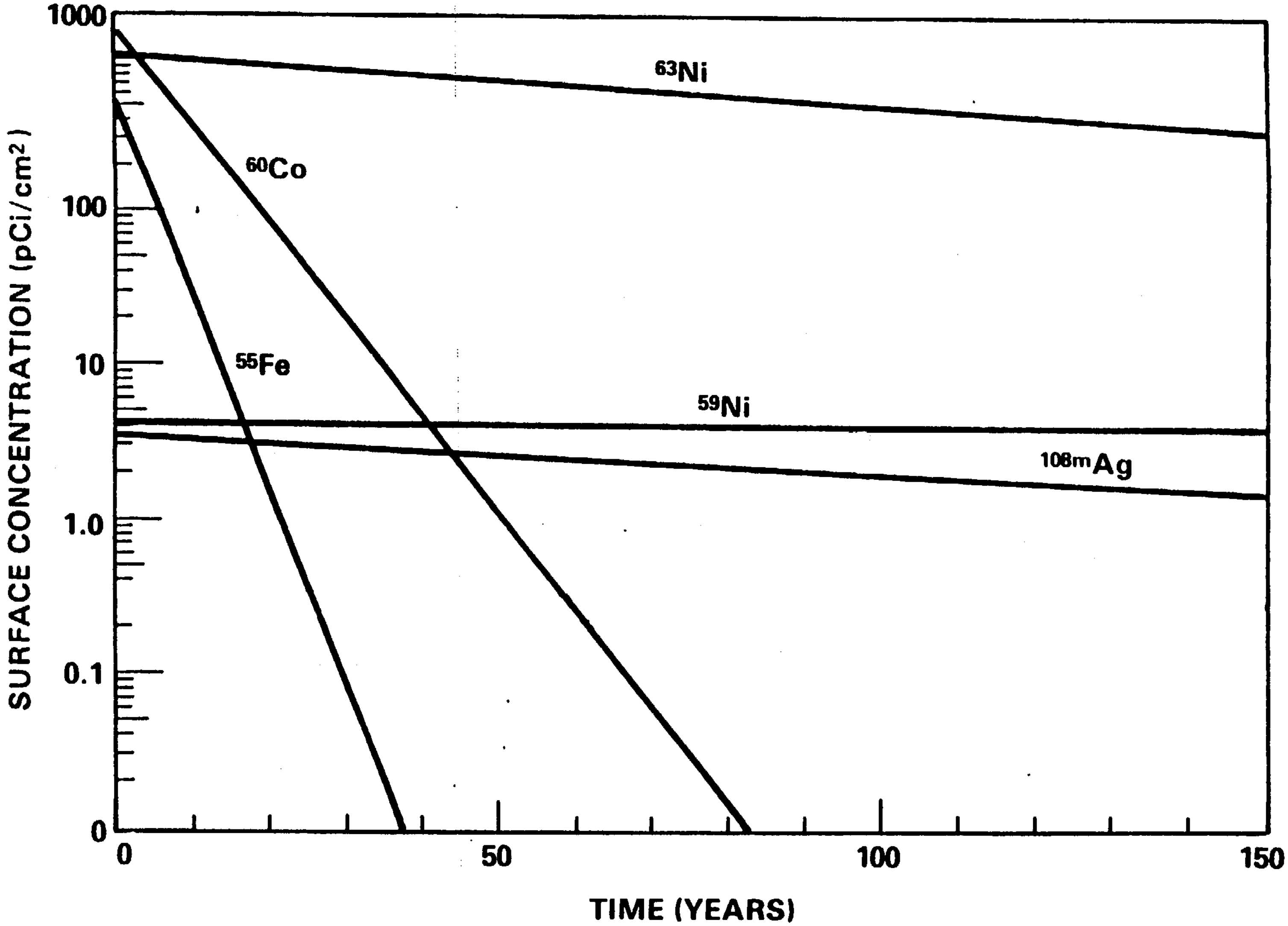


FIGURE 3.2. Change in Concentration With Time of Most Abundant Radionuclides in Main Steam Piping

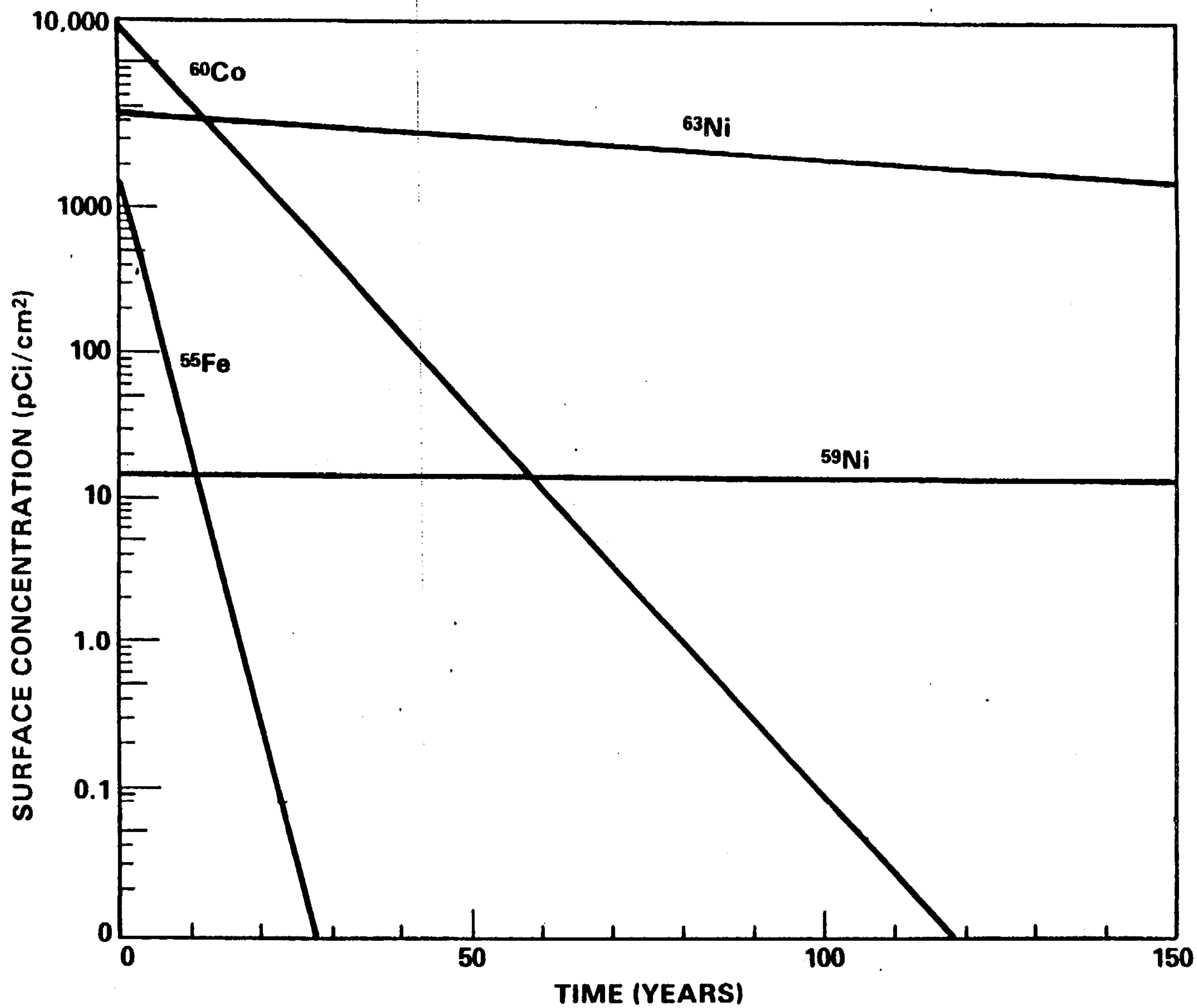


FIGURE 3.3. Change in Concentration With Time of Most Abundant Radionuclides in Reactor Feedwater Piping

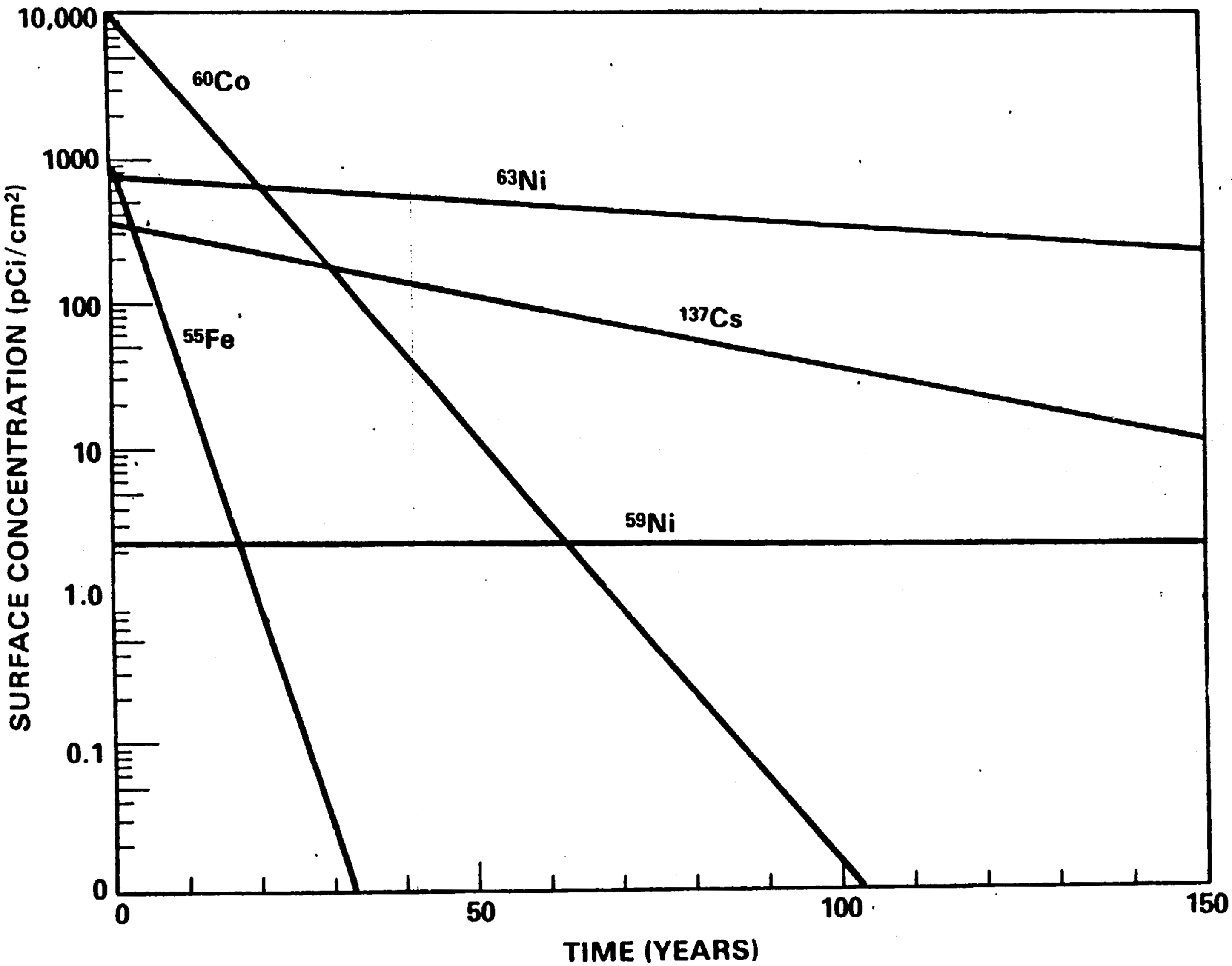


FIGURE 3.4. Change in Concentration With Time of Most Abundant Radionuclides in Reactor Purification Piping

The waste gas pressurizer tank (WGPT) and its associated piping contained only a trace of ^{137}Cs and ^{60}Co. Sample WGPT contained 2.8 pCi/cm^2 (0.27 pCi/g) of ^{137}Cs and 0.35 pCi/cm^2 (0.034 pCi/g) of ^{60}Co. These concentrations are so low that they are not even considered to be radioactive for the purposes of transporting or shipping. This radioactivity could not be detected by routine surveys with a G-M beta-gamma counter.

Likewise, the reactor building air vent ducts (RAVD) also contained only a trace of ^{60}Co and ^{137}Cs. Sample RAVD contained 0.89 pCi/cm^2 (0.67 pCi/g) of ^{60}Co and 0.028 pCi/cm^2 (0.021 pCi/g) of ^{137}Cs. Again, these concentrations are too low to be considered radioactive for transportation or shipping purposes.

Radiochemical Analysis of Selected Samples for ^{14}C, ^{99}Tc and ^{90}Sr

Although it was expected that no significant amounts of ^{14}C and ^{99}Tc would exist in the radioactivity residues at Pathfinder, several samples from important systems were radiochemically analyzed to verify its absence. The following samples indicated no significant amounts of these radionuclides were present:

TABLE 3.3. Carbon-14 and ^{99}Tc Concentrations in Pathfinder Piping

Sample	Concentration (pCi/cm^2) ^{14}C	^{99}Tc
Main Steam Line	1.6 ± 0.5	< 0.3
Reactor Feedwater Lines	< 0.3	< 0.3
Shield Pool Coolant Line	< 0.3	< 0.3
Reactor Feedwater Line	< 0.3	< 0.3

Concentrations of ^{90}Sr were also expected to be very low, and indeed they are, generally being lower than ^{137}Cs levels except for one sample, HSHT, the high solids holdup tank. As mentioned earlier, this pipe sample contained a residue of the solids contained in the high solids holdup tank and contained relatively high concentrations of all other radionuclides. The ^{90}Sr concentrations observed in a number of selected samples are given below.

Table 3.4. Strontium-90 Concentrations in Pathfinder Piping

Sample	^{90}Sr Concentration pCi/cm^2	(pCi/g)
Main Steam Line	<0.9	(<0.06)
Reactor Feedwater Line	<2	(<0.2)
Reactor Water Purification Line	<0.2	(<0.04)
Pool Water Clean-up Line	45	(18)
High Solids Holdup Tank	796	(325)
Fuel Storage Rack (Storage Basin)	0.20	(0.10)

The residual levels of ^{90}Sr throughout the plant appear to be negligible, except for the high solids holdup tank. The ^{90}Sr inventory in this tank would amount to approximately 170 microcuries, assuming that the sample collected was representative of the entire tank surface.

3.4.2 Concrete

Radionuclide concentrations observed in concrete cores collected at Pathfinder are presented in Tables 3.5 and 3.6. Table 3.5 contains the concentrations of the detectable radionuclides observed in the top 2 cm of the concrete cores. Most concrete samples had surprisingly low radionuclide concentrations, with ^{60}Co (the predominant radionuclide) usually ranging between 0.1 to 5 pCi/cm^2. However, several contaminated areas were pinpointed with a G-M counter and the concrete samples obtained from the locations contained above average concentrations of radioactivity. Two of these samples, PCC-1 and PCC-2, were collected from the floor of the former decontamination room in the fuel handling building. Cobalt-60 was the predominant radionuclide in sample PCC-2, and ^{137}Cs was predominant in PCC-1. Sample PCC-1 also contained ^{134}Cs (2.06 yr). The ^{137}Cs and ^{134}Cs were reportedly the result of contamination brought into the plant in contaminated fuel shipping casks which leaked some radioactivity on the floor during the offsite shipment of the spent Pathfinder fuel. This contaminated shipping cask was also the probable source of the ^{137}Cs observed on stainless steel structures in the fuel storage basin.

TABLE 3.5. Radionuclide Concentrations in Top Two Centimeters of Concrete Cores Collected at Pathfinder, September 1980

Sample	Surface Condition	Depth (cm)	pCi/cm² (a) ^{60}Co	^{134}Cs	^{137}Cs	^{152}Eu	^{154}Eu
PCC-1	clear sealer,	0-1	87.8	4.37	782.0	<0.04	<0.04
	rough surface	1-2	<0.07	<0.05	0.28	<0.03	<0.04
PCC-2	thick epoxy	0-1	353.0	<0.27	5.59	<0.23	<0.23
	coating	1-2	<0.02	<0.01	<0.01	<0.02	<0.01
PCC-3	clear sealer,	0-1	5.63	<0.01	0.054	0.032	<0.02
	smooth	1-2	<0.05	<0.05	<0.03	<0.03	<0.03
PCC-4	clear sealer,	0-1	1.49	<0.02	<0.03	0.036	<0.02
	smooth	1-2	<0.05	<0.05	<0.05	<0.03	<0.02
PCC-5	thick epoxy	0-1	2.12	<0.01	<0.02	<0.02	<0.02
	coating	1-2	<0.05	<0.05	<0.04	<0.02	<0.03
PCC-6	thick epoxy coating	0-1	3.72	<0.09	0.050	<0.03	<0.02
PCC-7	gray paint,	0-1	466.0	<0.08	0.29	0.27	<0.08
	smooth	1-2	8.90	<0.02	<0.02	0.086	0.031
PCC-8	chipped gray paint,	0-1	477.0	<0.08	0.46	1.33	0.23
	clear sealer	1-2	10.6	<0.02	<0.02	1.31	0.22
PCC-9A	chipped gray paint, clear sealer	0-1	350.0	<0.09	<0.36	<0.18	<0.09
PCC-9B	chipped gray paint, clear sealer	0-1	294.0	<0.01	0.39	0.33	<0.06
PCC-11	clear sealer, smooth	0-1	2.39	<0.01	0.072	<0.01	<0.02
PCC-12	clear sealer,	0-1	2.39	<0.01	0.072	<0.01	<0.02
	smooth	1-2	<0.03	<0.03	<0.03	<0.02	<0.02
PCC-13	clear sealer,	0-1	16.2	<0.01	<0.09	<0.02	<0.02
	rough surface	1-2	<0.05	<0.04	<0.04	<0.02	<0.03
PCC-14	gray paint,	0-1	0.14	<0.02	<0.06	<0.04	<0.02
	clear sealer	1-2	<0.02	<0.01	<0.01	<0.01	<0.01
PCC-15	thick epoxy	0-1	5.09	<0.01	0.17	<0.09	<0.02
	coating	1-2	<0.02	<0.01	<0.01	<0.01	<0.01
PCC-16	gray paint,	0-1	<0.045	<0.01	0.032	0.008	<0.02
	smooth	1-2	<0.05	<0.05	<0.04	<0.02	<0.03
PCC-17	gray paint,	0-1	0.98	<0.01	0.14	0.035	<0.02
	smooth	1-2	<0.06	<0.05	<0.04	<0.03	<0.03
PCC-18	gray paint, smooth	0-1	1.16	<0.01	<0.090	0.020	0.034
PCC-19	gray paint,	0-1	1.28	<0.01	0.045	0.01	<0.02
	smooth	1-2	<0.02	<0.02	<0.03	<0.01	<0.01
PCC-20	gray paint,	0-1	9.05	<0.02	1.91	0.47	<0.02
	smooth	1-2	<0.05	<0.05	<0.04	<0.02	<0.03
PCC-21	gray paint, smooth	0-1	0.39	<0.01	0.068	0.017	<0.02

(a) To convert to pCi/g, multiply pCi/cm² by 0.472

TABLE 3.6. Depth Distribution of Radionuclides in Concrete Cores Collected at Pathfinder, September 1980

		pCi/cm^2 (a)				
Core	Depth (cm)	^{60}Co	^{134}Cs	^{137}Cs	^{152}Eu	^{154}Eu
PCC-7	0-1	466	<0.08	0.29	0.272	<0.08
	1-2	8.90	<0.02	<0.02	0.086	0.031
	2-3	8.09	<0.01	<0.02	0.065	<0.01
	3-4	11.2	<0.01	<0.02	0.030	<0.01
	4-5	6.63	<0.01	<0.02	0.034	<0.01
	5-6	0.14	<0.009	<0.01	0.040	<0.008
	6-7	0.21	<0.01	<0.01	0.041	<0.009
	7-8	0.23	<0.009	<0.01	0.026	<0.009
	8-9	0.31	<0.009	<0.01	0.039	<0.009
	9-10	0.26	<0.009	<0.02	0.027	<0.009
	10-11	0.12	<0.009	<0.01	0.035	<0.008
	11-12	0.23	<0.01	<0.02	0.037	<0.009
PCC-8	0-1	447	<0.08	0.46	1.33	0.23
	1-2	10.6	<0.02	<0.02	1.31	0.22
	2-3	5.46	<0.01	<0.02	1.43	0.25
	3-4	5.61	<0.02	<0.02	1.50	0.22
	4-5	2.99	<0.02	<0.02	1.50	0.21
	5-6	3.05	<0.02	<0.02	1.90	0.26
	6-7	2.30	<0.01	<0.01	1.76	0.26
	7-8	2.45	<0.02	<0.02	1.85	0.27
	8-9	2.03	<0.02	<0.02	1.76	0.26

(a) To convert to pCi/g, multiply pCi/cm^2 by 0.472.

Concrete cores PCC-9A and PCC-9B were collected from the bottom of the steam chase in the reactor building and also contained above average residual radionuclide concentrations.

None of the cores (except PCC-7 and PCC-8 as noted below) contained any measurable radioactivity below a depth of one centimeter The concrete floors where the samples were taken were coated with either paint, an epoxy coating, or a clear cement sealer. All of these coatings helped prevent the spread of radioactivity with depth in the concrete. This is a very important observation since it seems that only the top cm of the most contaminated concrete at Pathfinder would need to be disposed of as radioactive material.

Another very important observation was that most of the radioactivity (^{60}Co and ^{137}Cs) could usually be removed from the concrete surface by stripping the paint or epoxy coatings. Paint and epoxy stripping was accomplished by brushing a coating of Zip-Strip® (a commercial paint remover) onto the painted surface, waiting 20 minutes, and then scraping off the softened paint with a metal blade. This worked very well for the gray paint and white epoxy used at Pathfinder. However, a clear sealer was also used at Pathfinder for coating the concrete floors, which appeared to be a silicate type of sealer. The Zip-Strip® paint remover had no effect on this clear sealer. As shown below, the removal of the paint surface was usually quite effective in removing most of the radioactivity from the concrete surfaces, except for those coated with the clear sealer.

The ^{60}Co was usually removed with greater efficiency compared to the ^{137}Cs. It is well known that ^{137}Cs will bind very tightly to concrete by an ion exchange mechanism, especially if the concrete contains significant amounts of clay and silicates.

In any case, it appears that most of the contaminated residue on concrete floors can be contained on the paint layer if a well maintained paint or epoxy coating is in place.

Cores PCC-7 and PCC-8 were collected from the pump floor in the reactor building. Core PCC-8 was taken right under the reactor pressure vessel and Core PCC-7 was collected near the middle reactor water circulation pump.

Table 3.7. Removal of Radionuclides from Concrete Surface by Stripping Paint Coatings

Concrete Core	Surface Condition	% Activity Removed by Stripping Paint ^{60}Co	^{137}Cs
PCC-2	Thick Epoxy Coating	99.8	80
PCC-3	Clear Sealer; Smooth Finish	0	*
PCC-6	Thick Epoxy Coating	96	100
PCC-7**	Gray Paint; Smooth Finish	68	0
PCC-9	Chipped Gray Paint; Clear Sealer	13	*
PCC-12	Clear Sealer; Smooth Finish	0	*
PCC-15	Thick Epoxy Coating	46	*
PCC-19	Gray Paint; Smooth Finish	77	*
PCC-20	Gray Paint; Smooth Finish	95	27

* ^{137}Cs concentrations too low to give an accurate number.
** This core is composed of mildly neutron activated concrete and contains some ^{60}Co incorporated in the concrete with depth.

These two cores are unique in that they have not only surface residual radionuclide contamination but also have been mildly neutron-activated and contain incorporated ^{60}Co, ^{152}Eu, and ^{154}Eu (see Figure 3.5). The depth distribution of ^{60}Co and ^{152}Eu in PCC-7 and PCC-8 is given in Table 3.4 and plotted graphically for PCC-7 in Figure 3.6 and for PCC-8 in Figure 3.7. The surface contamination (apparently due to spillage of reactor water on the floor) appears to have penetrated several cm into the concrete since the ^{60}Co decreased rapidly over the top 4 cm. Below about 4 cm, the ^{60}Co and ^{152}Eu which was neutron-activated in-situ is evident, and only slightly decreased with depth down to 12 cm. It was unfortunate that deeper cores were not taken at these locations. A more complete sampling of the concrete around and below the reactor pressure vessel would be necessary to precisely determine the radionuclide inventory of neutron activation products in this concrete.

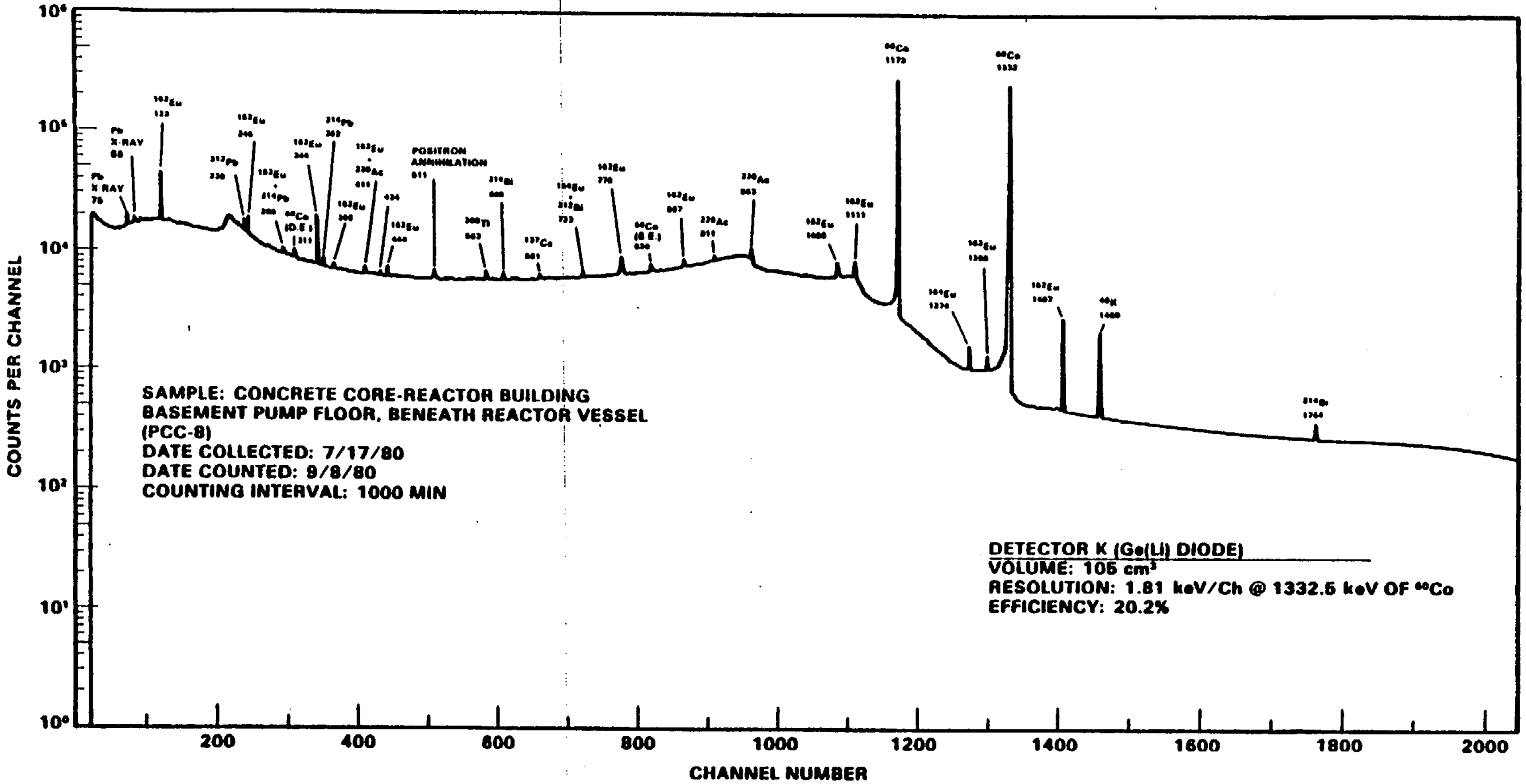


FIGURE 3.5. Gamma-Ray Spectrum of Concrete Core PCC-8 Collected Underneath Pressure Vessel

Since the concrete floor in the former decontamination area in the fuel handling building appeared to be randomly contaminated with small amounts of "fixed" radioactivity, a detailed survey of the distribution of these radionuclides was made in-situ using the portable intrinsic Ge gamma-ray spectrometer (see Figure 3.8). A total of 42 separate areas were analyzed by establishing a measurement grid of 6 x 7 locations two feet apart. A 10-cm-thick lead collimator shield was placed on the floor at each measurement spot and the intrinsic Ge detector was lowered into the collimator from a tripod stand and held 5 cm above the floor. Each spot was counted for 300 seconds. The results of this survey are shown in Figure 3.9. The only detectable radionuclides were ^{60}Co and ^{137}Cs, and the distribution of ^{60}Co and ^{137}Cs on the concrete floor was approximately 2 to 196 pCi/cm^2 and 2 to 175 pCi/cm^2, respectively. Three spots of above average surface contamination were observed where the ^{60}Co and ^{137}Cs concentrations ranged between approximately 82 to 196 pCi/cm^2 and 123 to 175 pCi/cm^2, respectively. These above average spots were believed to have originated from leaks from the fuel shipping casks during the offsite shipment of the Pathfinder spent fuel.

The radionuclide concentrations measured in all samples of Pathfinder piping, hardware, and concrete were orders of magnitude below the Class A waste classification category proposed in 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Wastes." Class A waste is waste that is segregated at the disposal site and disposed of with only minimal requirements on waste form and characteristics. Maximum radionuclide concentrations for Class A wastes have been published in 10 CFR 61 (USNRC, 1981), in units of $\mu Ci/cm^3$. If a density of 7.9 g/cm^3 for iron and steel, and 2.1 for Pathfinder concrete is used, the radionuclide concentrations listed in 10 CFR 61 can be converted into $\mu Ci/g$ of iron or steel and $\mu Ci/g$ of concrete for direct comparison with the concentrations measured in Pathfinder materials.

For making direct comparisons of the radionuclide concentrations in Pathfinder piping, hardware, and concrete with maximum allowable concentrations for Class A wastes, the data listed in 10 CFR 61 have also been converted into units of $\mu Ci/g$ of iron or steel and $\mu Ci/g$ of Pathfinder concrete (see Table 3.8).

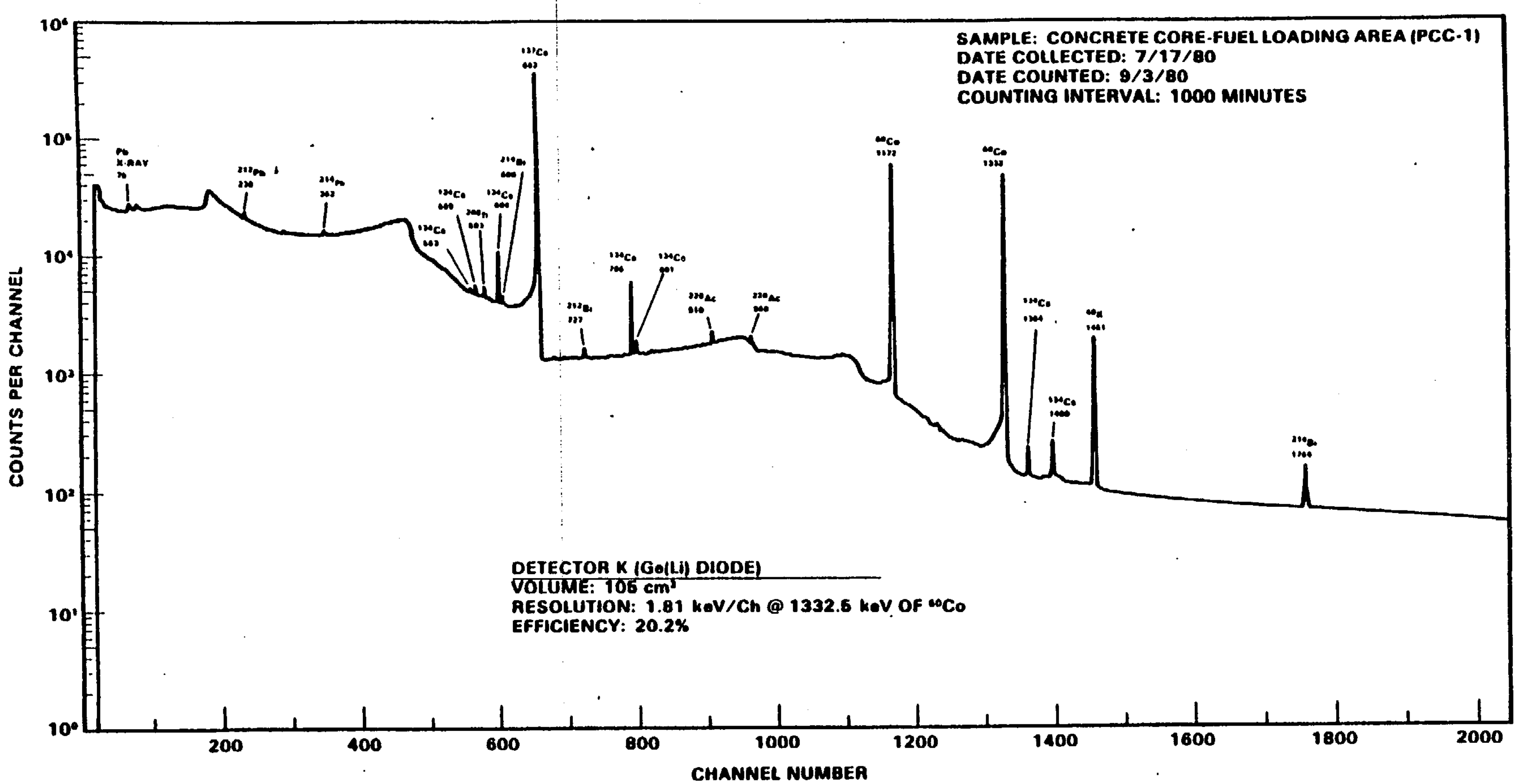


FIGURE 3.8. Gamma-Ray Spectrum of Concrete Core Samples from "Hot Spot" on Fuel Loading Area

NO. = ^{60}Co CONCENTRATION (pCi/cm^2)
(NO.) = ^{137}Cs CONCENTRATION (pCi/cm^2)

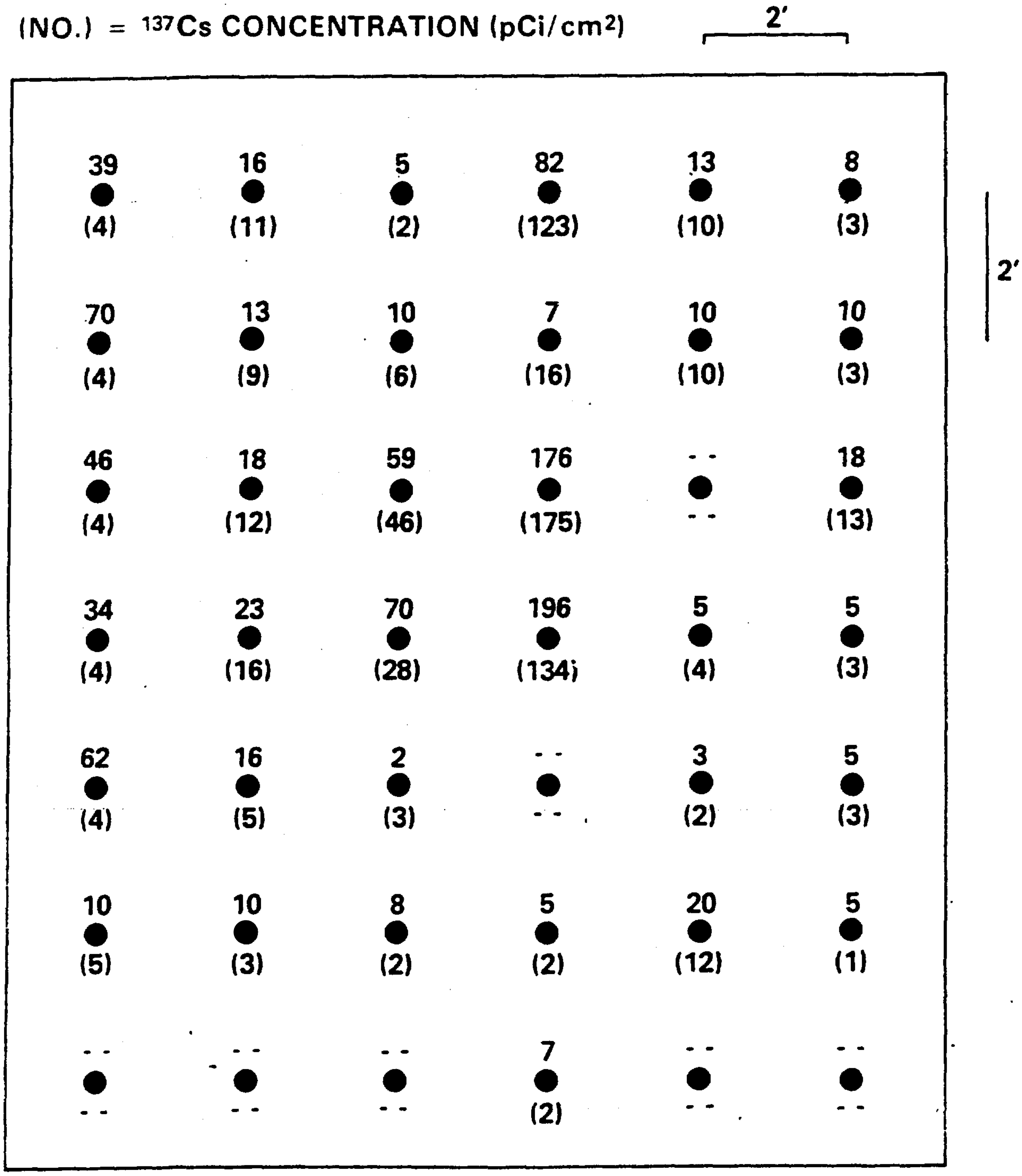


FIGURE 3.9. ^{60}Co and ^{137}Cs Distribution on Concrete Floor of Fuel Loading Dock

TABLE 3.8

Maximum Allowable Radionuclide Concentrations for Class A Segregated Waste

Radionuclide	$\mu Ci/cm^3$	$\mu Ci/gm$ of iron or steel[1]	$\mu Ci/gm$ concrete[2]
any with half life less than 5 years	700	89	333
H-3	40	5.1	19
C-14	0.8	0.10	0.38
Ni-59	2.2	0.28	1.0
Co-60	700	89	333
Ni-63	3.5	0.44	1.7
Nb-94	0.002	0.00025	0.00095
Sr-90	0.04	0.0051	0.019
Tc-99	0.3	0.038	0.14
I-129	0.008	0.0010	0.0038
Cs-135	84	10	40
Cs-137	1.0	0.13	0.48
Enriched U	0.04	0.0051	0.019
Natural or Depleted U	0.05	0.0063	0.024
Alpha Emitting transuranics	10 nCi/gm	10 nCi/gm	10 nCi/gm
Pu-241	350 nCi/gm	350 nCi/gm	350 nCi/gm

1 This column was derived by dividing the first column by the density of iron or steel (7.9 gm/cm^3)

2 This column was derived by dividing the first column by the density of Pathfinder concrete (2.1 gm/cm^3)

The highest ^{60}Co concentrations observed in Pathfinder piping (pool water clean-up line) was 0.0093 μCi/g, which is nearly four orders of magnitude below the Class A limit of 89 μCi/g of steel. The highest ^{63}Ni concentration in Pathfinder piping was about five orders of magnitude below the Class A limit of 0.28 μCi/g of steel The highest $^{239-240}Pu$ concentration observed in Pathfinder piping was about five orders of magnitude below the Class A limit of 10 nCi/g for alpha-emitting transuranic radionuclides.

The closest approach to the Class A limit was the ^{90}Sr concentration on steel piping from the high solids holdup tank. In this case, the observed concentration was about 16 times below the Class A limit of 0.0051 μCi/g of steel. The Pathfinder concrete likewise contained radionuclide concentrations which were many orders of magnitude below the Class A limits.

4.0 ESTIMATED RESIDUAL RADIONUCLIDE INVENTORIES

Radionuclide inventories presently residing in the major piping hardware systems in the Pathfinder nuclear plant have been estimated from concentrations measured in these systems.

4.1 PIPING AND HARDWARE

The data from Table 3.2 were used to develop estimates of the residual radionuclide inventories in the contaminated piping and hardware systems at Pathfinder. These inventories are listed in Table 4.1. The only detectable radionuclides observed in all systems at Pathfinder were ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni, ^{238}Pu, $^{239\ 240}Pu$, and ^{241}Am, with ^{108m}Ag, ^{137}Cs, and ^{152}Eu observed in a few samples. Therefore, the inventory estimates were made for only those radionuclides. Inventories in piping systems were estimated by examining engineering drawings and summing the total pipe lengths, computing a total inside surface area and multiplying this area by the radionuclide concentrations measured in the piping. In systems where no piping was obtained for analyses, the radionuclide inventories were estimated by using the radionuclide concentrations in similar piping which was measured. It was assumed for the inventory estimates that the radionuclide concentrations in the piping were homogeneously distributed on the inside surfaces, and that the measured samples were representative of the entire surface. This, however, was not the case in all systems because "hot spots" existed at bends, weld joints, pump connections, and other areas where the normal flow through the pipe was disturbed and where corrosion products could accumulate. Therefore, the inventories listed in Table 4.1 may be slightly underestimated. Also, some of the steam piping and reactor feedwater piping have been used in the fossil-fueled plant and have undoubtedly lost some of their radioactivity by desorption.

The radionuclide inventories in the empty tanks listed in Table 4.1 were estimated by multiplying the total inside surface area of the tanks by the radionuclide concentrations on the surfaces of piping which drained the tanks. This method could also give a slight underestimation of the total

TABLE 4.1. Radionuclide Inventories in Pathfinder Piping and Components, September 1980

System	Material Type	Nominal O.D. (in.)	Total Length (ft)	Total Weight (lb)	Total Inside Surface Area (cm^2)	Inventory of Detectible Radionuclides-Microcuries ^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{108m}Ag	^{137}Cs	^{152}Eu	^{238}Pu	$^{239-240}Pu$	^{241}Am
Main steam line (used MSL-A)	carbon steel	16	132	10,952	481,000	252	2.1	658	519	1.6	<0.21	1.2	0.00069	0.00075	0.00075*
	carbon steel	12	62	3318	180,000	94	0.7	246	194	0.6	<0.08	0.5	0.00026	0.00028	0.00028*
			TOTAL			346	2.8	904	713	2.2	<0.29	1.7	0.00095	0.00103	0.00103*
Steam emergency bypass line (used MSL-A)	carbon steel	16	11	910	40,100	39	0.33	227	82	0.14	0.003	0.018	0.00011	0.00012	0.00012*
	carbon steel	14	66	4,161	255,000	219	1.85	1,270	460	0.89	0.021	0.101	0.00062	0.00067	0.00067*
	carbon steel	12	19	1,021	55,400	54	0.45	373	113	0.19	0.005	0.025	0.00015	0.00017	0.00017*
	carbon steel	8	37	1,052	22,800	22	0.19	129	47	0.078	0.002	0.010	0.00006	0.00007	0.00007*
	carbon steel	4	22	231	21,000	20	0.17	119	43	0.072	0.002	0.009	0.00006	0.00006	0.00006*
			TOTAL			354	2.99	2,750	745	1.35	0.033	0.163	0.0010	0.00110	0.00110*
Reactor feedwater lines (used WP-42 all)	carbon steel	8	693	19,794	1,346,000	1,970	20	28,600	6,220	<0.100	<0.38	0.0013	0.0024	<0.0100	<0.010*
	carbon steel	8	115	2,194	170,000	250	26	3,610	790	<0.016	<0.013	<0.048	0.0002	0.0003	0.0003*
	carbon steel	4	100	1,075	97,500	140	15	2,070	450	<0.009	<0.007	<0.027	0.0001	0.0002	0.0002*
	carbon steel	3	218	1,652	163,000	240	24	3,460	750	<0.016	<0.012	<0.048	0.0002	0.0003	0.0003*
		TOTAL				2,600	85	37,740	8,210	<0.45	<0.13	<0.50	0.0018	0.0032	0.0032*
Reactor water purification lines (used WP-76-300-A3	stainless steel	6	544	1,024	79,700	74	0.18	1,691	368	<0.066	0.28	<0.33	0.000022	0.000025	0.00062
	stainless steel	4	90	967	87,800	81	0.20	1,863	405	<0.073	0.31	<0.37	0.000024	0.000028	0.00068
	stainless steel	3	575	2,488	428,000	395	0.99	9,082	1,977	<0.36	1.49	<1.8	0.000119	0.000139	0.00332
	stainless steel	2	43	159	21,900	20	0.05	465	101	<0.018	0.076	<0.09	0.000006	0.000007	0.00017
		TOTAL				570	1.38	13,101	2,851	<0.52	2.15	<2.6	0.000171	0.000199	0.0048
Reactor water purification demineralizer tanks (used WP-76-300-A3)	stainless steel				165,182	152	0.383	1,671	119	<0.14	0.57	<0.069	0.000046	0.000053	0.000053*
Reactor water purification coolers (used WP-76-300-A3)	stainless steel				165,560	153	0.384	1,675	119	<0.14	0.57	<0.069	0.000046	0.000053	0.000053*
Shield and storage pool purification lines (used SPD-A)	stainless steel	8"	4	99	6,792	2.3	0.004	31.2	1.18	<0.016	<0.058	<0.062	$<6\times10^{-6}$	$<4\times10^{-7}$	$<2\times10^{-7}$*
	stainless steel	4"	4	41	3,709	1.3	0.002	17.1	0.64	<0.008	<0.032	<0.034	$<3\times10^{-6}$	$<2\times10^{-7}$	$<1\times10^{-7}$*
	carbon steel	4"	161	1,737	157,612	442	0.23	123	75.7	<0.020	0.21	<0.082	0.0014	0.0089	0.010
	stainless steel	3"	192	1,455	143,184	48.4	0.082	659	24.8	<0.33	<1.23	<1.30	<0.00013	$<9\times10^{-6}$	$<4\times10^{-6}$*
Pool water cleanup line (PWC-8)	stainless steel	2"	36	131	18,092	46.0	0.097	427	29.8	<0.007	0.73	<0.030	0.00057	0.0038	0.0019
Inlet line to pool demin. (PDI)	stainless steel	2"	48	175	24,123	3.5*	0.0074*	32.5	2.3*	<0.006	0.031	<0.025	0.00004*	0.00029*	0.00014
Outlet from pool demin. (PDO)	stainless steel	2"	28	102	14,067	0.8*	0.0017*	7.5	0.53*	<0.0008	0.0034	<0.003	0.00001*	0.00007*	0.00004*
		TOTAL				544	0.424	1,297	135	<0.39	0.97	<1.5	0.0020	0.013	0.012
Shield pool (used FHR-A)	stainless steel	(26.5'I.D. 20' high)			2,060,300	113	<0.21	1623	72.9	<1.3	<1.7	<4.3	0.00066	0.0015	0.0062

* Radionuclide not actually measured in these samples. Inventory estimated by comparing with similar piping which was measured.

TABLE 4.3. Summary of Estimated Radionuclide Inventories in Pathfinder Plant Systems, September 1980

System	Inventory (millicuries)									
	^{55}Fe	^{60}Co	^{59}Ni	^{63}Ni	^{108m}Ag	^{137}Cs	^{152}Eu	^{238}Pu	$^{239-240}Pu$	^{241}Am(a)
Main steam and steam bypass lines	0.70	3.65	0.0058	1.46	0.036	<0.0003	0.0019	1.9×10^{-6}	2.1×10^{-6}	2.1×10^{-6} (a)
Reactor feedwater lines	2.60	37.7	0.085	8.21	<0.0005	<0.0001	<0.0005	1.8×10^{-6}	3.2×10^{-6}	3.2×10^{-6} (a)
Reactor water purification lines, coolers and tanks	0.88	16.4	0.0021	3.09	<0.0008	0.0033	<0.003	0.26×10^{-6}	0.31×10^{-6}	4.9×10^{-6}
Shield and storage pool purification lines, coolers and tanks	0.77	3.56	0.00091	0.28	<0.0005	0.001	<0.002	5.6×10^{-6}	30.3×10^{-6}	14.5×10^{-6}
Shield pool	0.11	1.62	<0.0002	0.073	<0.001	<0.002	<0.004	0.66×10^{-6}	1.5×10^{-6}	1.5×10^{-6} (a)
Fuel storage basin	2.16	32.4	0.0023	0.79	<0.009	0.083	<0.04	96×10^{-6}	13×10^{-6}	13×10^{-6} (a)
Rad-Waste piping and tanks	0.77	10.1	0.0034	0.37	<0.003	0.029	<0.01	700×10^{-6}	180×10^{-6}	180×10^{-6}
Total	7.9	105.4	0.10	14.2	0.036	0.12	0.0019	8.1×10^{-4}	2.3×10^{-4}	2.2×10^{-4}

(a) ^{241}Am inventories assumed to be equal to $^{239-240}Pu$ where measured values were not available.

Over 57% of the ^{63}Ni and 65% of the ^{59}Ni resided in the reactor feedwater piping. The reactor feedwater piping is carbon steel and the radioactive corrosion products were more efficiently adsorbed onto this surface compared to stainless steel. The main steam lines and the reactor purification lines were also important repositories for ^{59}Ni and ^{63}Ni.

The trace amounts of the transuranic radionuclides ^{238}Pu, $^{239-240}Pu$, and ^{241}Am resided mainly in the rad-waste piping and tanks; the total plant inventory for these radionuclides amounted to only 0.81, 0.23, and 0.22 microcuries, respectively.

4.2 CONCRETE

To arrive at an accurate estimate of the residual radionuclide inventory associated with the concrete surfaces of the plant would be very difficult because of the extreme variability in the radionuclide distribution on these surfaces. Residual contamination was very patchy, and within just a few feet surface radioactivity could vary by up to two orders of magnitude. Therefore, an estimate of the inventory of the residual radionuclide contamination of concrete surfaces was not attempted.

APPENDIX A

DESCRIPTION OF PATHFINDER GENERATING PLANT

APPENDIX A

DESCRIPTION OF PATHFINDER GENERATING PLANT

A.1 THE PHYSICAL PLANT

The Pathfinder Generating Plant was a 66 MWe boiling water type nuclear power station located 5.5 miles northeast of Sioux Falls, South Dakota (see Figure A.1). Northern States Power Company is the owner and operator. Construction of the nuclear plant was completed in early 1964 and initial criticality was achieved on March 24, 1964. The plant operated intermittently through a period of 42 months, and in September 1967 was shut down due to failure of the steam separators within the reactor vessel. It was then decided, for economic reasons, to terminate the nuclear operations and convert the plant to a gas/oil fired unit. The retrofitted fossil-fueled plant is still in use today, but only during periods of peak power demand. The present appearance of the plant is shown in Figure A.2.

A schematic drawing of the reactor and auxiliary buildings is shown in Figure A.3, and general information describing the important operating parameters is given in Table A.1. A brief description of the reactor and plant components and systems is given to provide information relating to the origin of various radionuclides and their transport pathways through the plant.

A.1.1 Reactor and External Components

The reactor complex consists of the reactor vessel and its internals, the recirculation pumps, valves and piping and the control rod drives (see Figure A.4). The reactor pressure vessel is fabricated of stainless steel-clad carbon steel, and housed the nuclear fuel, associated support structures and internal components required to produce superheated steam. Three recirculation pumps forced water at temperatures approaching the boiling point through the boiler fuel elements at a total rate of about 65,000 gpm. Steam, in an amount equal to steam demand, was boiled from the recirculation water in the boiler core, sent to the turbine generator, condensed and then returned to the reactor as feedwater.

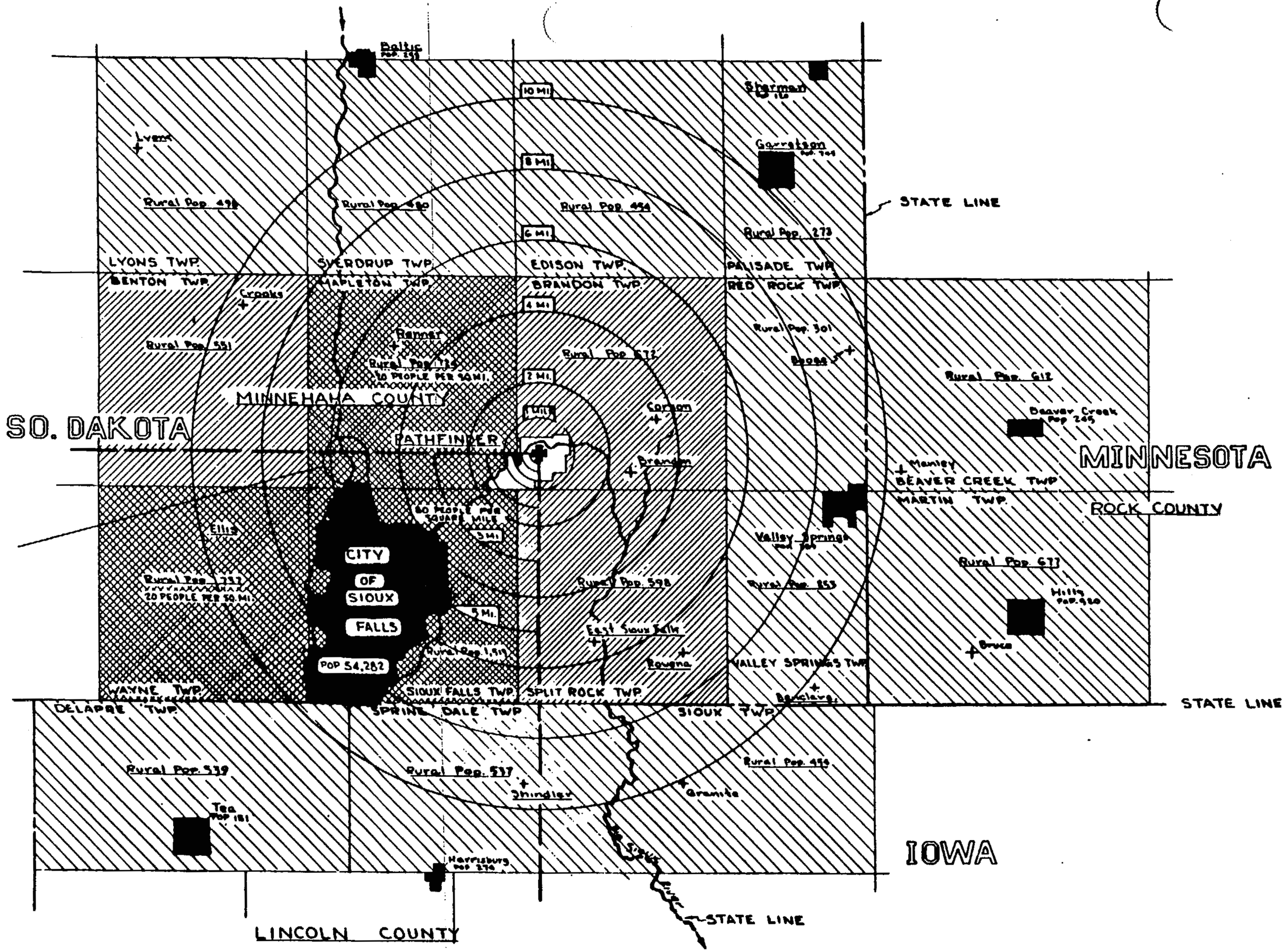


FIGURE A.1. Location of Pathfinder Generating Plant

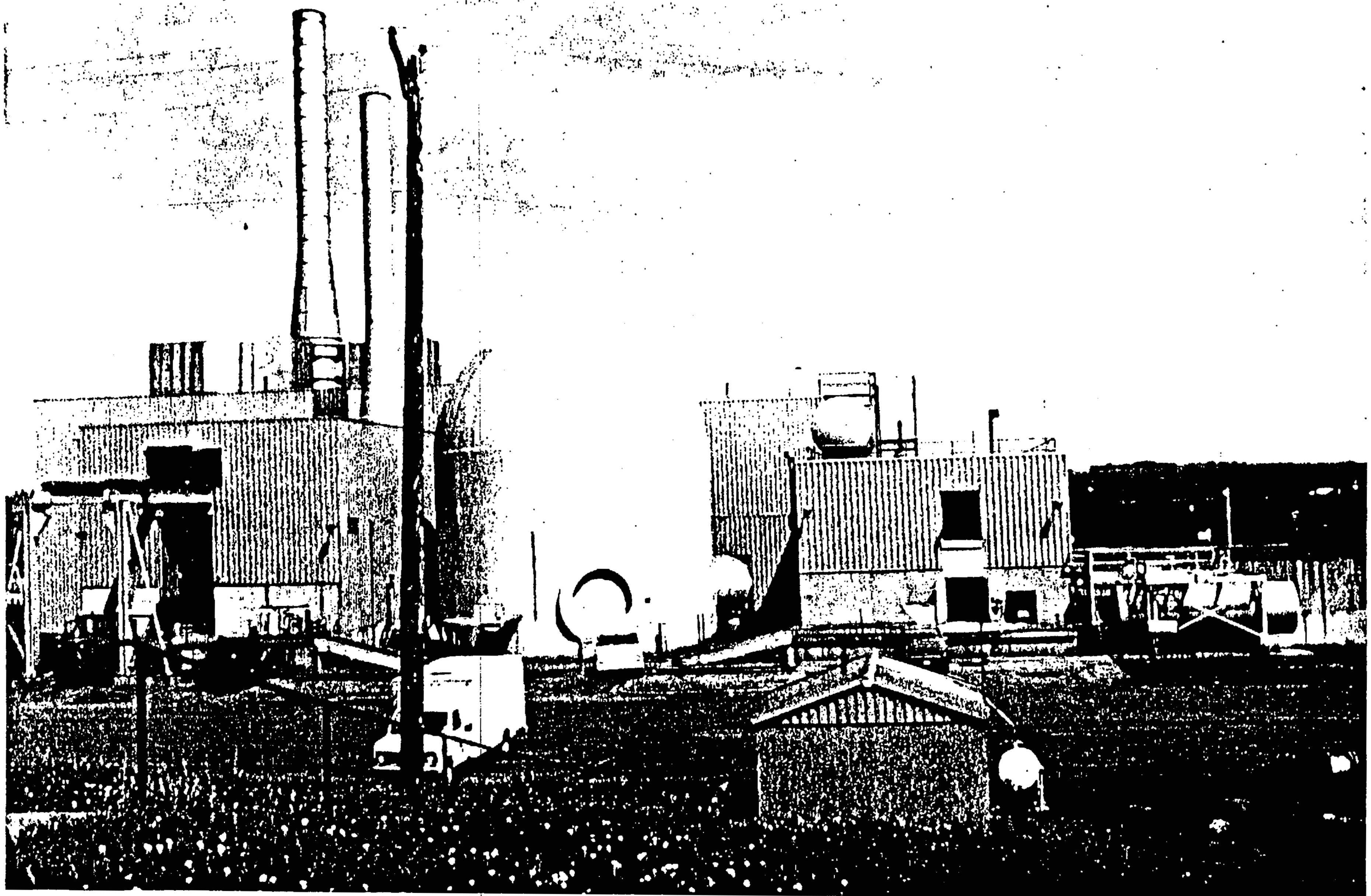


FIGURE A.2. Pathfinder Generating Plant, Sioux Falls, South Dakota

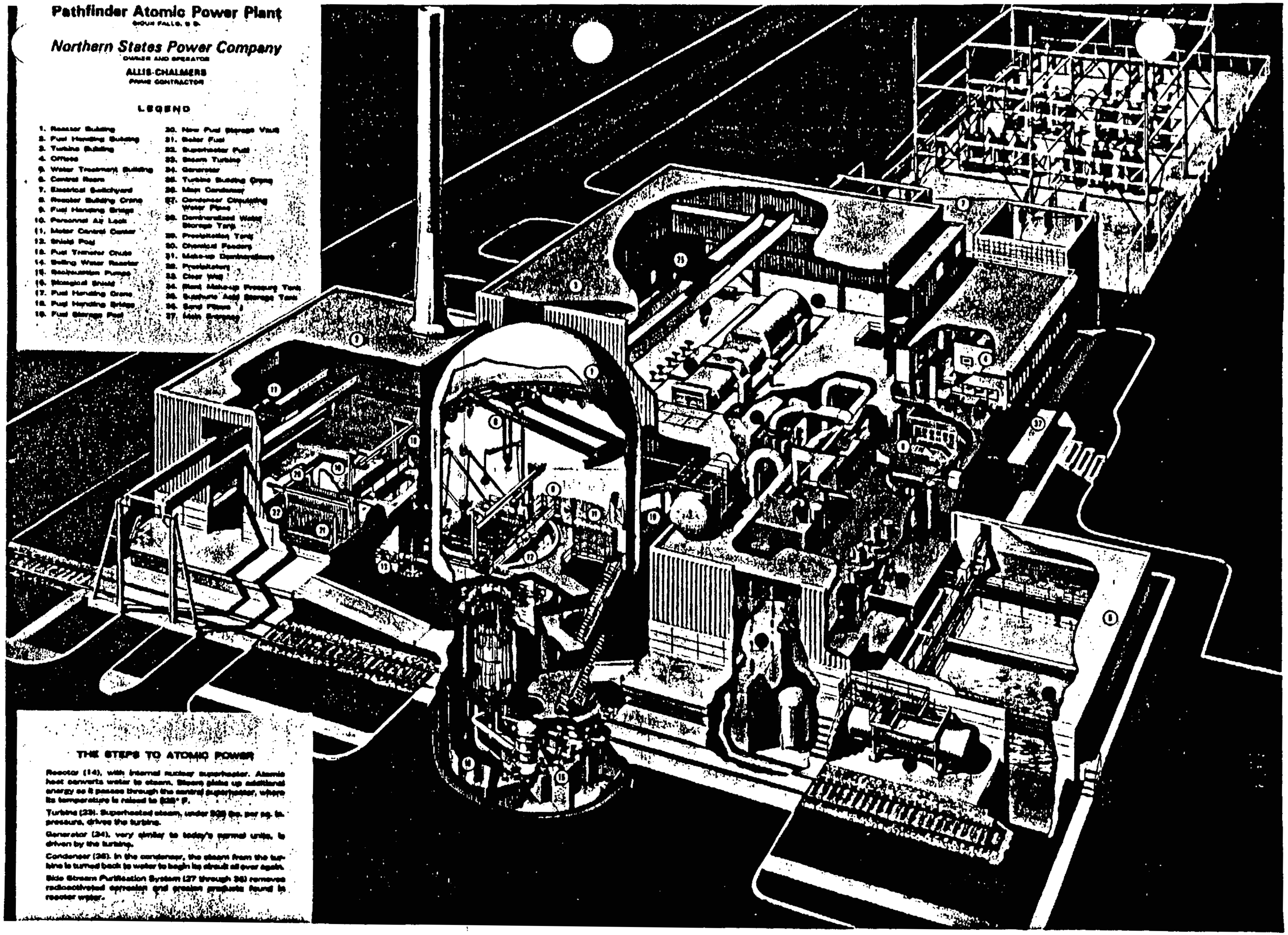


FIGURE A.3. Schematic Drawing of Pathfinder Generating Plant

TABLE A.1. Pathfinder Generating Plant

General Information

Location--	5.5 miles northeast of the center of Sioux Falls, South Dakota
Reactor Type--	Controlled recirculation boiling water
Coolant-Moderator--	Light water
License Number--	DPR-11
Prime Contractor--	Northern States Power Company
Principal Subcontractor--	Allis-Chalmers (development, design, & construction)
Architect/Engineer--	Pioneer Service & Engineering, Inc.

Participants in Research and Development--
Central Utilities Atomic Power Association (CUAPA)-

Central Electric & Gas Company	Northern States Power Company
Interstate Power Company	Northwestern Public Service Company
Iowa Power & Light Company	Otter Tail Power Company
Iowa Southern Utilities Company	St. Joseph Light & Power Company
Madison Gas & Electric Company	Wisconsin Public Service Corporation

Boiler Region Heat Output--	164 MWt	Net Thermal Efficiency--	30.5%
Superheater Region Heat Output--	39.7 MWt	Reactor Outlet Pressure--	540 psi
Heat Output Total--	203 MWt	Reactor Outlet Temperature--	825°F
Station Output (Gross)--	66 MWe	Reactor Operation Pressure--	600 psi
Station Output (Net)--	62 MWe	Steam Generation Rate--	6000,000 lbM/hr
Coolant Inlet Velocity:			
Boiler Region	14.2 fps		
Superheater Region	99 fps		

TABLE A.1. (contd)

Coolant Outlet Velocity:	
Superheater Region	116 fps
Average Heat Flux--	
Boiler Region	131,000 Btu/hr ft^2
Superheater Region	85,000 Btu/hr ft^2
Estimated Burnout Heat Flux-- Boiler	1×10^6 Btu/hr ft^2
Maximum Heat Flux--	
Superheater	208,000 Btu/hr ft^2
Core Dimensions--	
Cylindrical	72 in.
Diameter--	
Central Superheater	30 in.
Both	72 in. high
Fuel Element Configuration--	
Boiler	92 subassemblies of square lattice bundles of fuel rods
Fuel Type--	
Boiler	1.8% enriched UO_2 pellets with Zircaloy-2 cladding
Superheater	93.0% enriched UO_2 cermet with Type 316L stainless steel cladding
Maximum Fuel Temperature--	
Boiler	4800°F
Maximum Surface Temperature--	
Superheater	1300°F
Average Exit Void Fraction--	
Boiler	42.4%
Average Exit Quality--	
Boiler Region	2.558%
Forced Recirculation Ratio--	39.1
Feedwater Temperature--	360°F

TABLE A.1. (contd)

Control Rods--	
Boiler Region	16 rods--10 in., cruciform-shaped, made of 1/4-in.-thick 2% boron stainless steel
Superheater Region	4 rods--9 in., cruciform shape
Pressure Drop--	
Boiler Region	18.1 psi
Superheater Region	55.0 psi
Operation Period--	March 1964 through September 1967
Personnel--	10 Professional Personnel 21 Operating Personnel 5 Health Physicists and Chemists 16 Other 52 Total

Cost of Pathfinder:

Capital Costs	$25,772,731.83
Total Cost	42,088,980.36
Participants' Cost	
NSP	30,457,000.25
CUAPA	3,650,000.00
Total--	$34,107.000.25

Mothballing & Conversion

Dates--	May 1967 through November 1971
Total Cost--	$1,868,571.00
Object of--	Convert to possession-only license
Present Status--	Fossil-fueled generating plant, used only for peak periods

Comments: Pathfinder was shut down in September 1967 due to the failure of the steam separaters within the reactor vessel. It was then decided, for economic reasons, to terminate nuclear operations and convert to a gas/oil fired steam supply system.

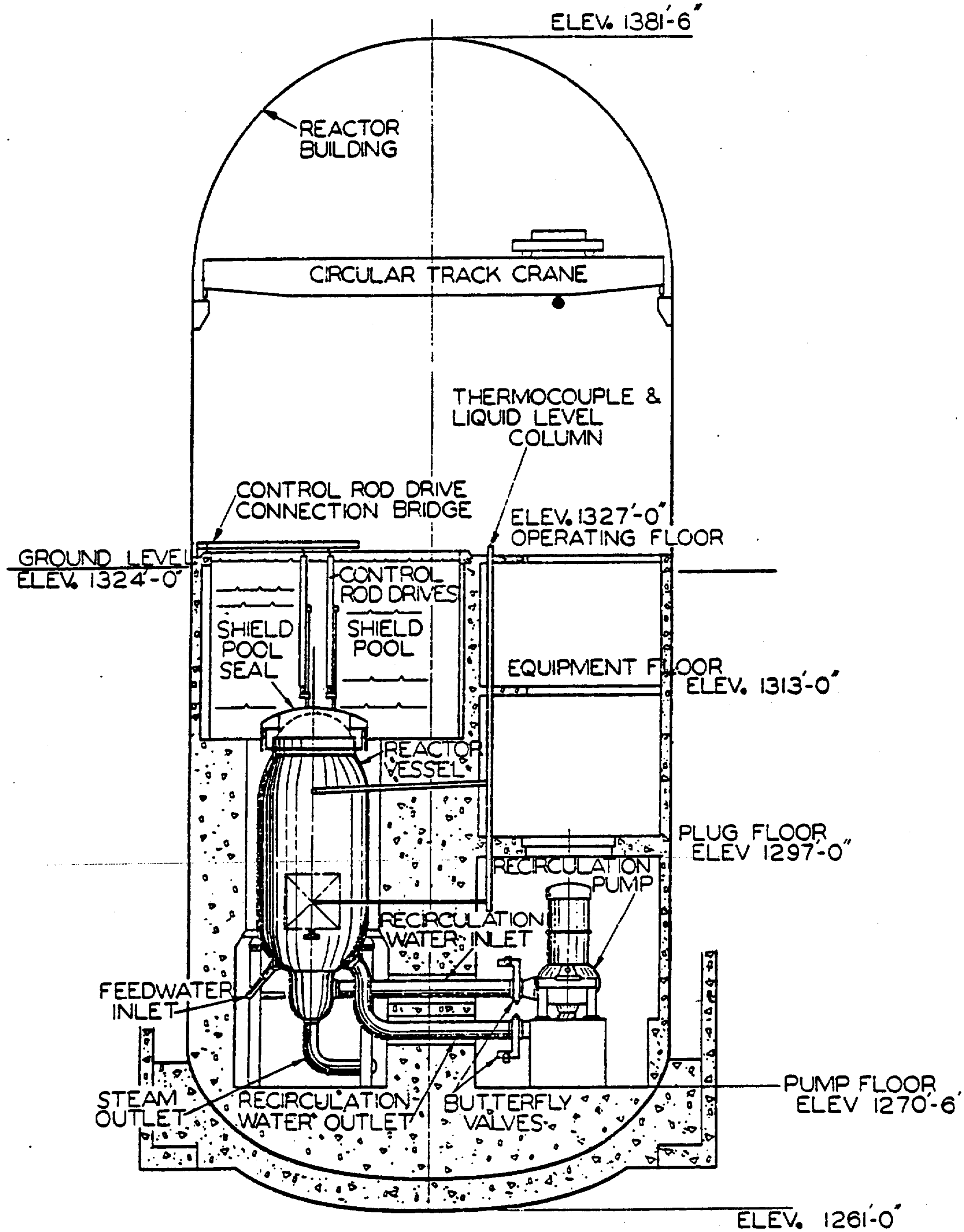


FIGURE A.4. Reactor Building

The main steam line outlet was through a 16-in. nominal diameter carbon steel pipe designed to conduct the superheated steam from the bottom of the reactor vessel directly to the steam turbine. Condensed steam was returned to the reactor pressure vessel through an 8-in. nominal diameter carbon steel pipe connected to the feedwater pumps.

The control rod drives of the reactor were mounted on the reactor pressure vessel head and operated in the water shield pool.

A.1.2. Reactor Internal Components

Although the scope of this study does not include an assessment of the radionuclide inventory in the neutron-activated pressure vessel and internal components, a brief discussion of these structures is included here because the design and composition of the reactor internals directly influenced the radionuclide mixture which was generated and translocated to other systems of the power plant.

The Pathfinder reactor was a heterogeneous boiling water type with a two-region core--a boiler region and a superheater region (see Figure A.5). The boiler region of the reactor core consisted of 96 fuel element assemblies (1.8% enriched UO_2) approximately 5 in. square by 99 in. long overall. The fuel elements in the boiler region were clad with Zircaloy-2, an alloy composed of 1.5% tin, 0.1% iron, 0.1% chromium, 0.05% nickel, 0.035% carbon, and the balance being zirconium. The boiler region fuel produced saturated steam from the recirculation water. Water served as both a coolant and a moderator.

The superheater region of the reactor core resembled a tube-type heat exchanger consisting of 415 fuel-bearing tubes (93.0% enriched UO_2) approximately 1 in. in diameter. The superheater fuel elements were clad with Type 316L stainless steel. Saturated steam, produced in the boiler region, flowed through the annular elements, was superheated to 725°F, and exited at the bottom of the pressure vessel directly to the steam turbine.

Most of the other internal components of the pressure vessel, including the steam separators, boiler baffle, boiler grid plate and various support structures were composed of either 304 or 304L stainless steel (2% Mn, 18% to

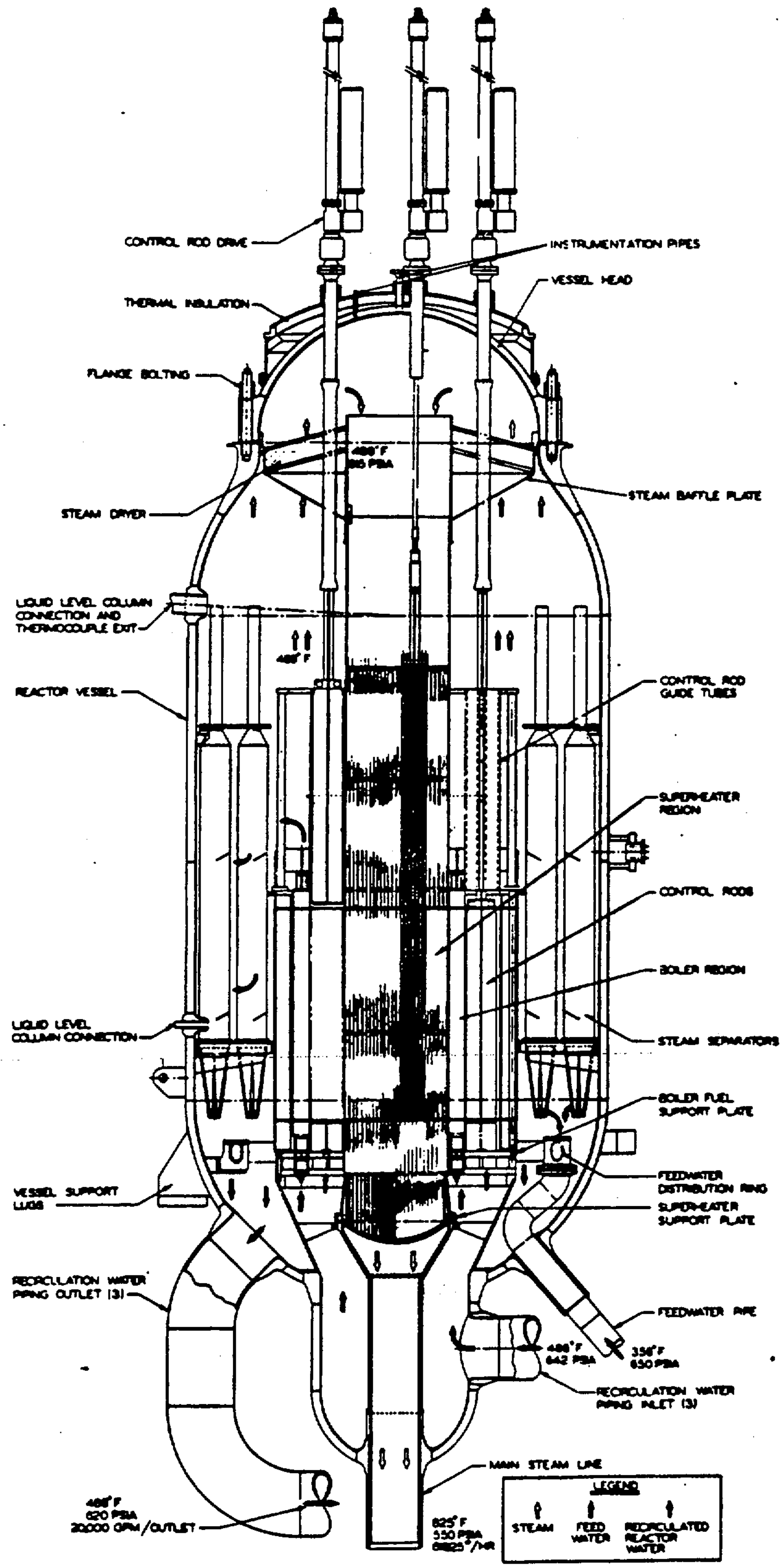


FIGURE A.5. Controlled Recirculation Boiling Reactor (CRBR) with Nuclear Superheater for Pathfinder Generating Plant

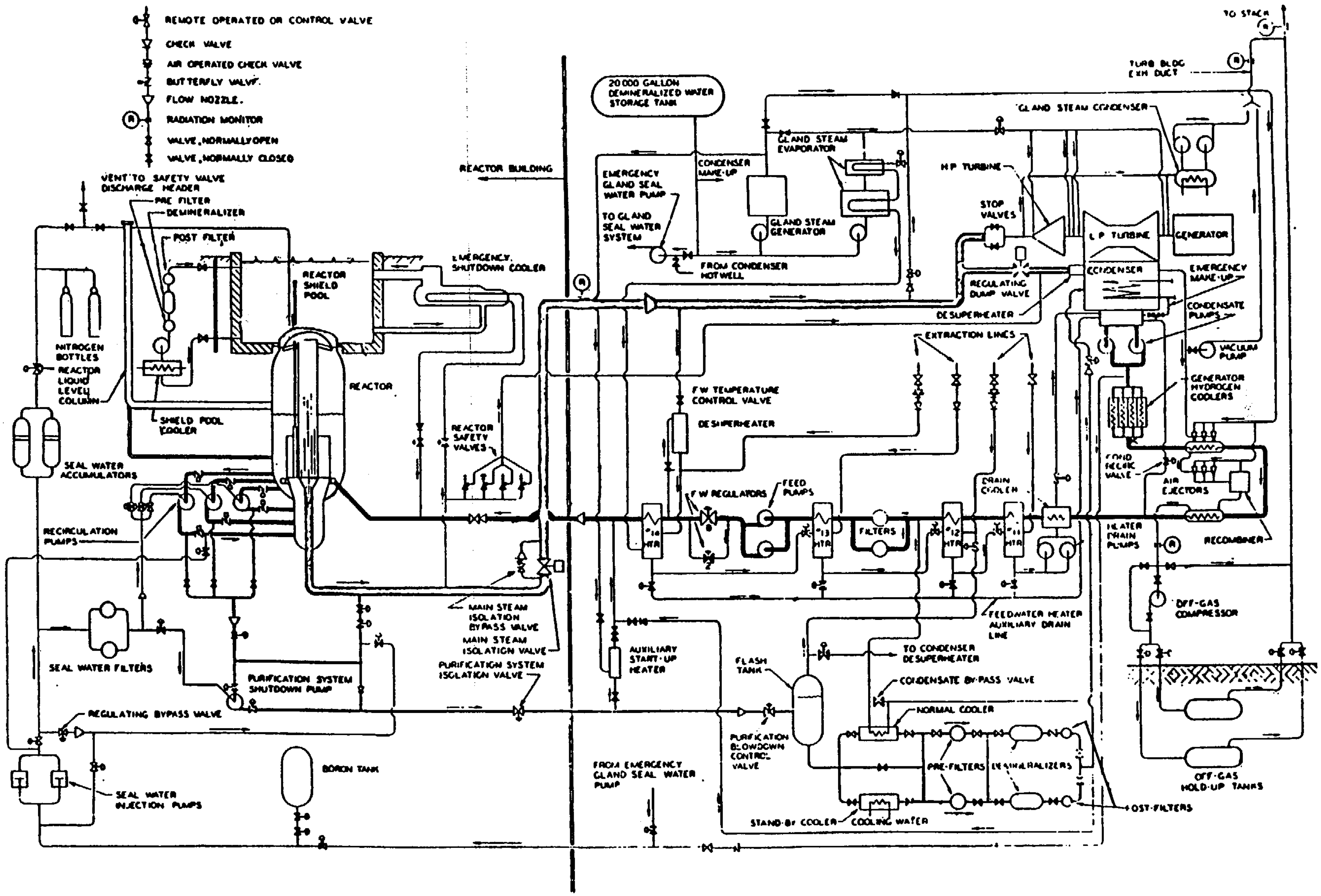


FIGURE A.6. Pathfinder System Components and Flow Diagram

20% Cr, 8% to 12% Ni). Only the steam dryer assembly was composed of inconel wire mesh (76% Ni, 16% Cr, 7% Fe 0.2% Mn).

Control of the reactor power level was maintained by adjustment of control rods located in both the boiler and superheater region of the reactor. Rack and pinion type control rod drives were mounted on top of the pressure vessel and were employed to position the control rods in the nuclear core to effect a change in power level. The drives also forced the control rods into the core in the case of an emergency shutdown. The neutron absorbing section of the control rods was composed of borated (2%) 304 stainless steel.

The feedwater was returned to the vessel and injected into the recirculation water through a feedwater distribution ring located above the recirculation water outlet pipes, thus insuring a maximum amount of mixing with the recirculation water.

A.1.3 Steam Line

The main steam line was 16-in. Schedule 60 pipe fabricated from A-335 Grade P II steel. [illegible] line was equipped with a motor-operated isolation valve (see Figure [illegible]. This valve, located within the reactor building, was tripped on scrams [illegible]ing isolation to maintain the integrity of the reactor building. It [illegible]d as a main steam line stop valve during normal startup and shutdo[illegible]ations. A 4-in. line, bypassing the isolation valve, provided for [illegible] flows up to 60,000 pounds/hr, and also had an isolation valve. The ma[illegible]team line divided into two 12-in. lines at the turbine.

A 12-in. main steam dump line was connected to the main steam line ahead of the turbine trip valves. It was used to bypass steam around the turbine to the desuperheater pipe located in the condenser. The dump valve installed in this line automatically controlled the steam line pressure when the pressure was not controlled by the turbine inlet valves. The valve was also used during plant startup and shutdown operations.

A.1.4 Turbine

The turbine was a 66 MWe tandem-compound, double-flow 3600-rpm unit utilizing convention design standards and materials of construction. Necessary modifications to the convention turbine shaft seals were made to eliminate the possibility of reactor steam escaping into the turbine building.

The steam conditions at the turbine inlet valves for initial operation of the first core were 525 psig and 722°F with an estimated flow of 500,000 pounds/hr which produced 50 MWe. For test periods during first core operation, reactor steam flow could be increased to approximately 616,125 pounds/hr. The turbine supplied steam to four regenerative feedwater heaters from extraction points in the turbine casing (see Figure A.6).

A.1.5 Condenser

The two-pass surface condenser was designed for a heat load of 435,000,000 Btu/hr at a vacuum of 28-1/2 in. Hg. With Admiralty metal tubes (70% Cu, 29% Zn, 1% Sn) and a deaerating hotwell, it was of standard design with the following exceptions:

- The hotwell was sized to provide storage for the water used to flood the reactor steam dome and superheater during shutdown.
- When steam bypassed around the turbine or relieved through the reactor safety valves, it entered the condenser through the desuperheater pipe (see Figure A.6).

Noncondensable gases were removed from the condenser by a triple element, two-stage steam jet air ejector. A rotary vacuum pump was also provided for use during startup and shutdown.

The condenser was cooled by circulation water which flowed to the condenser by gravity from the basin of a convention cross-flow, induced draft cooling tower. The water was returned to the cooling tower by two circulating water pumps which took suction from the condenser outlet water box.

The cooling tower water make-up treatment system treated water pumped from the Big Sioux River. The system consisted of two river pumps, two traveling screens, two precipitators, a clearwell and two cooling tower water make-up pumps.

A.1.6 Feedwater System

The feedwater system included two condensate pumps, a drain cooler, two condensate filters, four regenerative feedwater heaters, two feed pumps and a feedwater control valve (see Figure A.6). Three low pressure heaters were located between the condensate pumps and the feedwater pumps. One high pressure heater, utilizing reactor steam, was located between the feedwater pumps and the reactor. Each of the above pumps and filters was sized for full normal capacity flow, thus one set was available for stand-by service. Bypass lines were provided for each feedwater heater.

Make-up water was supplied from the 20,000 gallon demineralized water storage tank to the condenser hotwell.

A.1.7 Emergency Shutdown Cooling System

The emergency shutdown condenser used shield pool water to condense reactor steam following a scram requiring isolation. Reactor steam flowed to the tube side of the condenser through a 6-in. line which was connected to the main steam line at a point located between the reactor steam outlet and the main steam line isolation valves (see Figure A.6). The steam was condensed in the emergency shutdown condenser; the condensate flowed by gravity through a 3-in. line into the reactor feedwater line and returned to the reactor.

Cooling water was supplied to the shell side of the emergency condenser from the reactor shield pool through an 8-in. pipe. Steam generated in the condenser was returned by natural convection to the shield pool through a 20-in. pipe. There was sufficient heat sink capacity in the shield pool water to remove reactor decay heat for approximately 24 hours. Heat could be removed from the pool water or make-up water could be added continuously to the pool by means of the shield pool cooling system if prolonged isolation was necessary.

A.1.8 Reactor Water Purification System

In order to maintain high purity reactor coolant, a side stream low with the philosophy of incorporating as many secondary functions as practical. The primary function of the purification system was to remove activated corrosion products normally found in the reactor water. This was necessary to keep system radioactivity levels low, minimize the possibility of plateout on heat transfer surfaces and the accumulation of activated crud in parts of the system where it could make direct maintenance difficult or impossible. During startups the purification system also served as a means of reactor pressure control, and during shutdowns it also served as a secondary reactor water cooling loop.

The system was designed to maintain a total solids concentration in reactor water of less than 1 part per million (ppm). During normal operation, up to 110 gpm of reactor water was fed to the purification system.

During the reactor startup procedure, the superheater and a section of the main steam line were drained through the purification system.

Most of the components of the purification system, coming in contact with reactor water, are made of stainless steel. The piping system is of welded construction.

During normal plant operation, reactor water was bled from the discharge nozzles of the three reactor water recirculation pumps. The water flowed to a common header where the combined flow was up to 100 gpm. It then flowed through a high pressure line which penetrated the reactor building. The water flowed to the fuel handling building where all of the purification system components were located.

At this point, the high pressure reactor water was throttled into a flash tank which was maintained at approximately 35 psig by controlling steam flow directly to the No. 12 feedwater heater. The flash tank steam flow was condensed in this feedwater and returned to the feedwater system. This steam could be bypassed to the main condenser if necessary (see Figure A.6).

Approximately 62% of the flow left the flash tank as saturated water at flash tank pressure. After leaving the flash tank, the water was cooled to 140°F or less in a cooler, which used condensate as the normal coolant. From there, it flowed through a system of prefilters, ion exchangers, and after-strainers to the main condenser. All piping, filter housings and tanks were constructed of stainless steel.

The flash tank is located in the fuel handling building. It was housed in a separate concrete vault. The tank, mounted vertically on the mezzanine floor, measured 34 in. in depth and 7 ft, 6 in. high and was entirely of 304 stainless steel.

There are two shell-and-tube coolers located in a shielded area in the fuel handling building. One was used as the normal cooler, and the other as a standby and shutdown cooler.

Water from the flash tank flowed through a heat exchanger type cooler where the purification water temperature was reduced to 140 F or less by condensate.

The secondary cooler was used during phases of reactor startup and shutdown. It also acted as a standby to be used in place of the normal cooler if necessary. Its capacity was greater than that of the normal cooler, enabling it to be used for reactor shutdown cooling. Water from the cooling tower system was used as coolant.

Two 100-ft^2, element-type precoat filters preceded the ion exchangers. During normal operation one filter was used with the other on standby. This type of filter was remotely precoated and backwashed, thereby minimizing exposure of personnel by activated filter crud.

The mixed bed ion exchangers removed the soluble corrosion products from the reactor water, and stopped any suspended particles which could have passed through the prefilters. Each unit was designed for a 125-gpm maximum flow. For limited periods the ion exchangers could be operated in parallel at a combined flow of 250 gpm. Underdrains for resin retention consisted of metal edge type strainers. Resin was charged and removed from the vessels by hydraulic sluicing.

An additional small strainer was used following each ion exchanger. This strainer prevented resin from entering the condenser in the event of an ion exchanger underdrain failure.

A.1.9. Radioactive Solid Waste Disposal System

Solid Wastes

Solid wastes were composed of combustible wastes, noncombustible wastes, filter cartridges, filter aid material, spent ion-exchange resins, used fuel elements, fuel element boxes and control rods. Table A.2 gives an estimate of the expected quantities of solid waste materials which were generated. Disposal consisted of packaging and shipping offsite for burial at a commercial low-level waste disposal site.

Combustible wastes, consisting of rags, paper, clothing, mops, rubber goods, as well as noncombustible wastes such as nuts, bolts, gaskets, glass and small parts of equipment, were collected, monitored for radioactivity, and packaged in special shielded containers. These containers were shipped offsite for ultimate disposal.

Filter cartridges that were radioactive were transferred from the filter body into a suitable container by semiremote methods. After storage to permit some radiation decay, these containers were shipped offsite for ultimate disposal. Filter aid material, backwashed from filters, was sluiced to a 2,000-gallon concentrated waste storage tank. This material, after concentrating by sedimentation, was stored in the 12,000-gallon spent resin tank. After a decay period, the wastes were placed in suitable containers for offsite shipment.

Spent ion-exchange resins from the reactor water purification, pools purification, and liquid waste systems were transferred by sluicing to a shielded 12,000-gallon stainless steel spent resin storage tank which was located underground adjacent to the fuel handling building. The tank provided capacity for storing resin from 3 to 5 years of operation. The resin,

TABLE A.2. Estimated Quantities of Solid Radioactive Wastes

Combustible Wastes	10,000 lb/yr.
Noncombustible Wastes	2,000 lb/yr.
Filter Cartridges	1,000 lb/yr.
Spent Resins	350 cu ft/yr.
Filter Backwash Materials	1,000 lb/yr.

after approximately a 4-year decay period for reduction in radioactivity, was pumped to shielded shipping containers. These containers were shipped off-site for ultimate disposal.

Used fuel elements, fuel element boxes and control rods were stored in the fuel storage pool for a decay period. They were then transferred to a shielded shipping coffin. Used fuel was shipped offsite to a fuel processing plant. Radioactive metal parts were shipped to an authorized area for disposal offsite.

A.1.10. Radioactive Liquid Waste Disposal System

General Concept

The function of the liquid waste system was to monitor and, if necessary, to process radioactive water generated in the plant and to discharge these liquid wastes from the site within the maximum permissible concentrations established by state and Federal Regulations or to make liquids suitable for reuse within the plant. The major design criteria for the system were to prevent any radiation hazard either to operating personnel or the public.

The methods that were used for reducing radioactivity in wastes were:

Decay
Filtration
Demineralization
Sedimentation
Dilution

Whenever practical, processed water was returned to the system for reuse. Processed water that was not suitable or needed for reuse was disposed of after dilution with cooling tower blowdown water, by discharging to the Big Sioux River via a ditch. Radioactive water not suitable for processing was prepared for shipment offsite. Liquid waste and cooling tower blowdown discharged to the ditch were controlled to insure that the radionuclide concentrations did not exceed the limits of 10 CFR 20, Standard for Protection Against Radiation, January 1, 1961. Regulation of cooling tower blowdown and waste discharge flows provided the necessary dilution for liquid waste discharged from the plant. Blowdown flow as little as 310 gpm provided sufficient dilution for the approximatey 4 x 10^{-6} μc/cc activity in the 11,000 gallons of liquid waste generated each day.

System Description

Liquid wastes were generated from the following sources and amounted to about 11,000 gallons per day with activity of approximately 4 x 10^{-6} μc/cc.

Sources of Liquid Wastes

Equipment Drains

- Reactor Water
- Condensate
- Water Drained from Equipment
- Gland Sealing Drains
- Building Drains

Spent Resin Transport Water

- Reactor Water Purification System
- Pools Cleanup System
- Waste Disposal System

Filter Backwash Material Transport Water

- Reactor Water Purification Precoat Filters
- Pools Cleanup Filter Backwash
- Waste Disposal Precoat Filters

Chemical Drains

- Decontamination Solutions
- Cleaning Solutions
- Laboratory Sink Drains
- Sampling Sink Drains

Liquid wastes from these sources were accumulated in sumps. Liquids in the sumps were pumped to the waste disposal system for process or discharged to the ditch for dilution. Table A.3 lists the estimated volumes and activity of liquid wastes generated during normal reactor plant operation as well as during the maintenance and refueling period. Refer to Figure A.7 for a

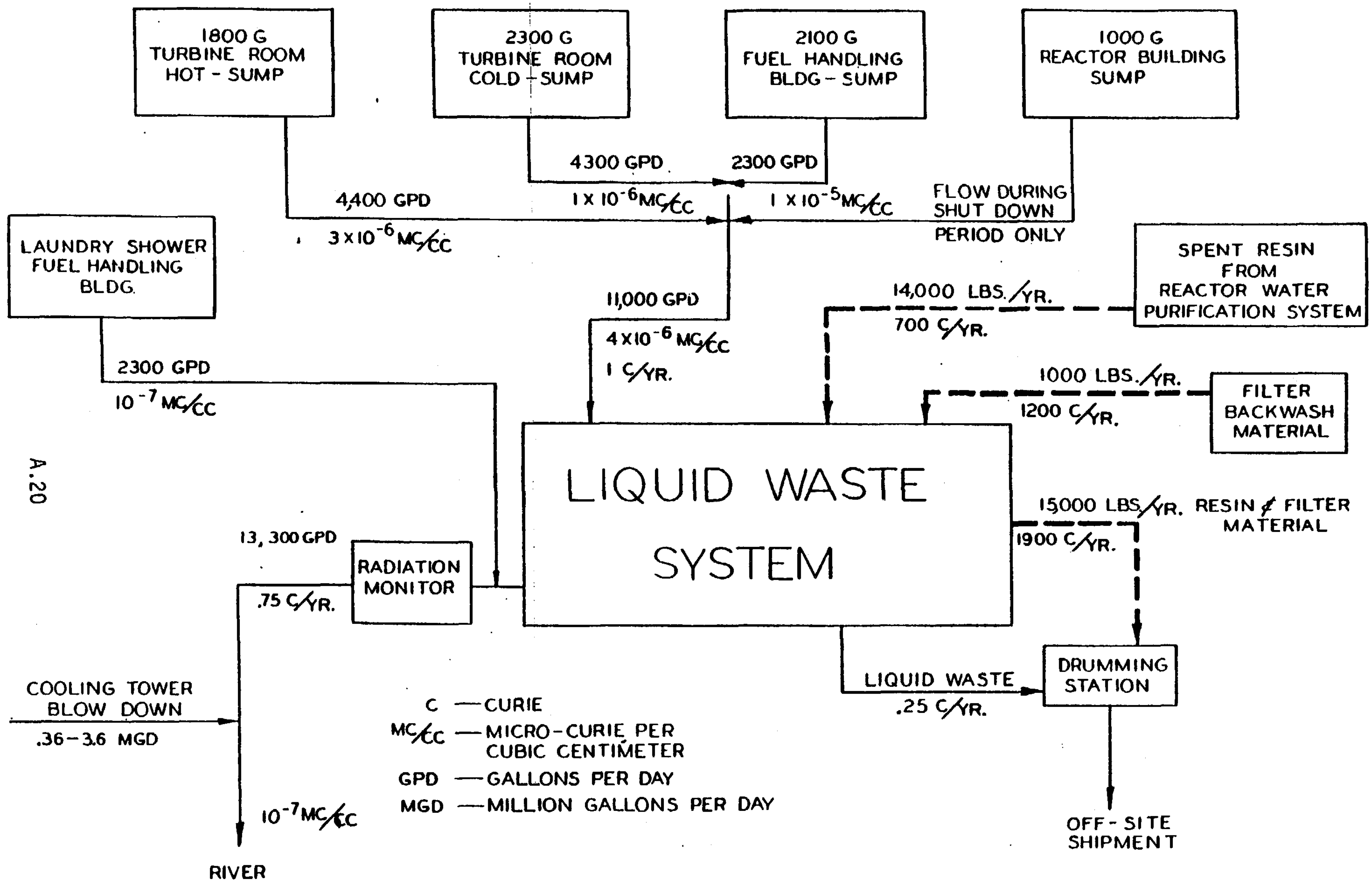


FIGURE A.7. Liquid Waste Production and Handling at Pathfinder simplified diagram showing collection and discharge of liquid wastes with approximate associated activity.

The system had a storage capacity for more than two days' production of liquid waste. Wastes produced each day were processed and discharged at about the same rate as the wastes were generated. Thus, the short-term storage capacity of 23,200 gallons shown in Table A.4 served as a dynamic storage or surge capacity. Long-term storage capacity refers to 17,800 gallons provided in tanks where liquid wastes were stored for a few months to several years. These tanks contained high activity as well as high solids content until they were prepared for offsite shipment.

The system was operated on a batch basis which provided for close control of all processing. A simplified flow scheme is presented in Figure A. . Each batch of waste was sampled to determine its radioactivity and chemical composition. After sampling, it was processed in one of the following ways:

- Liquid waste with activity of $<10^{-3}$ - $<10^{-5}$ μc/cc was processed by filtration and demineralization to a level of 10^{-5} - 10^{-6} μc/cc. Liquid waste having activity of 10^{-5} or 10^{-6} was discharged with dilution.
- Low activity waste (10^{-5} - 10^{-6} μc/cc) was discharged to the ditch in such a manner as to control the final activity of the wastes in the ditch water at 10^{-7} μc/cc activity level above background.

TABLE A.3. Liquid Waste Entering System

Sump Location	Normal Plant Operation (10 Months Per Year) gpd	μc/cc	Shutdown Period (2 Months Per Year) gpd	μc/cc
Turbine Room--Hot	4,400	3×10^{-6}	200	1×10^{-4}
Turbine Room--Cold	4,300	1×10^{-6}	---	---
Fuel Handling Bldg.	2,300	1×10^{-5}	400	8×10^{-3}
Reactor Building	---	---	7,400	1×10^{-5}
	11,000	4×10^{-6}	8,000	4×10^{-4}

TABLE A.4. Waste Disposal System Storage Capacity

Tanks	Gallons Short Term	Gallons Long Term
Low Solids Hold-Up	12,000	
Waste Surge	1,000	
Reclaimed Water	3,000	
Neutralizing Hold-Up		300
Neutralizing Tank		1,500
High Solids Hold-Up		2,000
Concentrated Waste Storage		2,000
Spent Resin		12,000
Collection Sumps		
Turbine Room--Cold	2,300	
Turbine Room--Hot	1,700	
Fuel Handling Building	2,200	
Reactor Building	1,000	
	23,200	17,800

- High activity (10^{-2} - 10^{-3} μc/cc) low solids (ionic content) water was processed by filtration and demineralization. Each batch was checked for radioactivity after processing and before it was released for reuse or discharge to the ditch. If an acceptable activity level (10^{-5} - 10^{-6} μc/cc) was reached by one-cycle processing, the waste was recycled to reduce activity.

- High activity, high solids wastes were neutralized, to facilitate processing. Then after a decay period, when an acceptable low activity level had been reached, it was discharged to the ditch for dilution. If discharge was not feasible, these wastes were stored for further decay and concentration by settling before being shipped off-site for disposal.

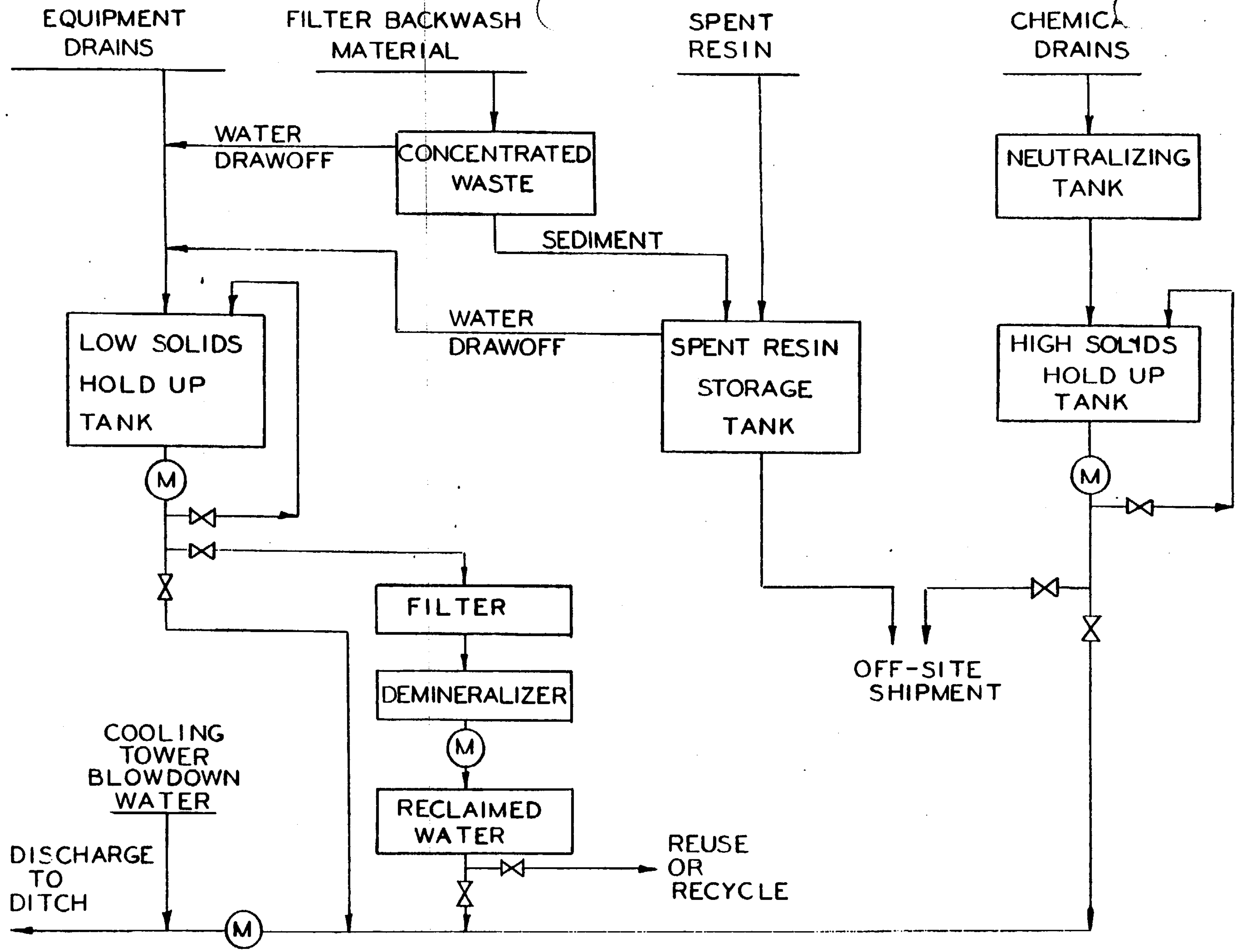


FIGURE A.8. Liquid Waste Disposal Simplified Flow Scheme

Control and Monitoring

Water with activity <10^{-3} μc/cc was sent to the low solids hold-up tank and separated into the four compartments according to ionic content and activity level ranges.

A.1.11. Gaseous Waste Disposal System

General Concept

During operation of the plant, there were three sources of potentially radioactive gaseous contamination.

1. Gaseous radioisotopes produced by neutron activation of primary water.
2. Gaseous fission products.
3. Gaseous radioisotopes produced through activation of shield cooling air.

Argon-41 and nitrogen-16 were the most prevalent radioisotopes from activation, whereas xenon and krypton were the principal gaseous fission products. With perfect fuel elements there would be no xenon and krypton released since it would be contained by the cladding. However, it is recognized that:

- Small amounts of U-235 can become impregnated into the external surfaces of the cladding. This U-235 cannot be removed by any practical or economic method.
- There may be minute diffusion of fission gas through the cladding.
- Cladding failures might occur which would permit leakage of fission gas from the fuel elements.

In order to reduce the activity level of the radioactive gases produced in the plant, the following methods were used:

- Decay
- Filtration
- Dilution
- Atmospheric Dispersion

System Description

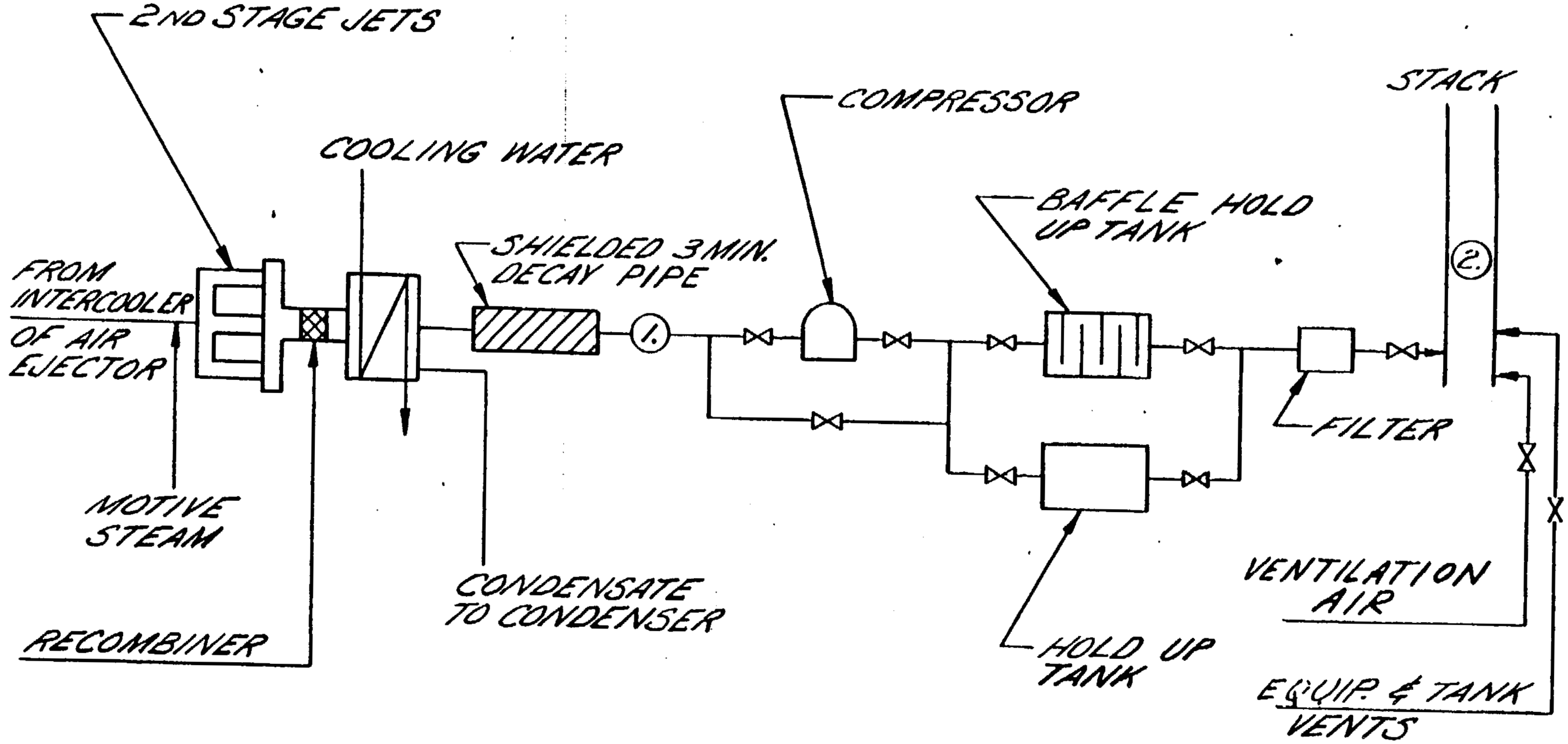
The noble fission gases and other noncondensables from the reactor would appear in the main steam. Any traces of the halogens and nonvolatiles carried by the steam were stripped out in the condenser and remained in the condensate. The noncondensables were removed from the condenser by the air ejector system (see Figure A.6). A catalyst bed located between the inter and after coolers of the air ejector was used to recombine the oxygen and hydrogen from dissociated reactor water. The remaining noncondensables flowed to the radioactive gaseous waste disposal system.

Figure A.9 shows the basic flow diagram for this system. Following the air ejectors was a three minute hold-up pipe to permit decay of the short-lived nuclides such as ^{16}N, ^{17}N, ^{19}O, ^{90}Kr, ^{139}Xe and ^{140}Xe. The gas then flowed through a baffled hold-up tank giving an additional 12-minute delay before being discharged through an absolute filter to the stack. Here the radioactive gas was diluted with ventilation air before being discharged up through the stack.

In the event of a major release of radioactive gas, the activity would be too high for discharge to the stack. If this were the case, the gas was compressed into one or both of two hold-up tanks. Each tank had sufficient capacity for approximately 12 hours of full power operation. After a decay period, the gas in the tanks would be released to the stack in such a manner as to remain within acceptable limits. It should be mentioned that during the 42-month operation of the Pathfinder nuclear plant no significant release of fission product gases occurred.

The ventilation air discharged to the stack contained ^{41}Ar resulting from irradiation of ventilation air in the reactor biological shield. This activity was diluted with other building ventilation air in the stack and dispersed to the atmosphere. This insured that the concentration of ^{41}Ar in unrestricted areas was below the specified maximum permissible concentration for air.

Vent gases from the waste disposal system were fed to a common header which entered the stack plenum. They were diluted with ventilation air in the stack and also dispersed to the atmosphere.



(1.) HIGH ACTIVITY ALARM MONITOR

(2.) CONTINUOUSLY RECORDING MONITOR

FIGURE A.9. Gaseous Waste Production and Handling at Pathfinder

APPENDIX B

NUCLEAR PLANT OPERATING HISTORY

APPENDIX B

NUCLEAR PLANT OPERATING HISTORY

The brief operating history of the Pathfinder reactor (42 months) was organized into four phases. The first three phases of the initial start-up program consisted of a series of tests designed to demonstrate the physics performance of the reactor and the performance of the integrated plant operation. These tests were sequenced in their order of increasing reactor power level, i.e., all tests at a given power level were performed before the power was raised to the next higher level. Phase IV was to consist of routine operations at full power. However, Phase IV was not activated due to equipment failure and for economic reasons. These phases and their operational histories relating to this study are summarized as follows.

B.1 Phase I--200 kW(TH) OR LESS--MARCH 1964 TO FEBRUARY 1966

Phase I experiments were performed at low power and ambient conditions to establish the reference core. Critical control rod configurations, core power distributions, and reactivity coefficients were measured for the reference core.

Between March 24, 1964 (date of initial criticality) and November 9, 1964, boiler fuel core criticality tests were performed. On November 9, 1964, the superheater fuel was loaded and on November 16, 1964, initial criticality for the full core was achieved. Reactor and integrated systems performance tests during Phase I continued through February 1966.

During Phase I, the radioactivity generated by the reactor was minimal. No fuel element failures occurred, and radioactivity levels in the reactor water were kept very low (ranging from 5×10^{-8} to 1.1×10^{-5} μCi/mℓ and averaging 2.3×10^{-7} μCi/mℓ). No significant contamination problems with either airborne or smearable contamination were encountered. No abnormal or unexpected exposures to personnel occurred. Gaseous radwastes released on an unidentified basis during Phase I amounted to 2.5×10^{9} μCi of noble gases and 8.3×10^{4} μCi of particulates. The concentrations released were below MPC levels. The noble gases and particulates were believed to be composed mainly of naturally-occurring radionuclides of the uranium and thorium chains originating from concrete and construction materials. Periodic analysis showed that all activity indicated by the inline monitors was background or naturally-occurring particulates. No halogen radionuclides were released. Liquid radwastes discharged to a ditch which drained to the Big Sioux River amounted to 4.7×10^3 μCi during Phase I.

B.2 PHASE II--5 MW(TH) OR LESS--MARCH 6 TO MAY 19, 1966

The objective of the Phase II testing was to raise the reactor power level from essentially zero power to a level at which the onset of boiling in the flooded superheater was expected, and to determine the superheater radiative cooling ability, perform reactor systems tests and calibration of nuclear instruments. During Phase II, the integated reactor thermal power achieved was 4.68 MWD. The superheater fuel burnup for this period was equivalent to 6.05 MWD/MTU, and the boiler fuel burnup was 0.655 MWD/MTU.

The radioactivity levels in the reactor water substantially increased during this period, ranging from 1.5×10^{-7} to 9.2×10^{-4} μCi/mℓ and averaging 1.7×10^{-4} μCi/mℓ. No halogen activity was observed in the reactor water indicating that no fuel element failures had occurred. Gaseous radwastes released during Phase II were at background or naturally-occurring levels and amounted to 3.0×10^8 μCi of noble gases and 6.5×10^3 μCi of particulates. Liquid radwaste discharges to the Big Sioux River via the drainage ditch amounted to 1.7×10^2 μCi.

No significant contamination problems with either airborne or smearable activity were encountered during Phase II, and no abnormal or unexpected exposures occurred.

B.3 PHASE III--FULL POWER OR LESS--MAY 19, 1966 TO SEPTEMBER 16, 1967

The objective of this phase was to reach full power in a safe manner. During Phase III, power was increased in about five steps, starting from some power near 5 MW(TH) and going to full power. At each power level numerous tests were performed, including power calibration, radiation testing, superheater steam operation, xenon reactivity, fluid dynamics effects, and a variety of systems testing.

Phase III lasted for a period of 17 months, ending on September 16, 1967, when the plant was shut down due to a condenser tube failure. Subsequent inspections showed that the steam separators had begun to disintegrate near the nozzles and activated pieces of translocated metal were found in various locations within and outside the reactor pressure vessel. Because of this hardware failure problem and also due to economic reasons, the reactor was never restarted.

The Phase III testing had been quite successful, and between May 19, 1966, and September 16, 1967, the reactor produced 16,621 MWD and logged 4595 critical hours. The total boiler fuel burnup was approximately 2330 MWD/T.

During this period, the radioactivity levels in the reactor water increased to an average concentration of 7.4 x 10^{-2} μCi/mℓ of gross beta activity, with levels reaching as high as 2 to 3 μCi/mℓ. No fission products were observed in the primary system indicating no fuel element failures. Radioactive corrosion products began to deposit in various piping loops to the point where some shielding of hot spots was required. High dose rate areas included the purification line in the fuel handling building, the air ejector, feed water heaters, the main steam line, pool clean-up equipment, seal water filters, chloride analyzer filters, spent resin tanks, the purification cooler system, and various pumps. The radionuclides reported to be present in fresh primary samples included ^{64}Cu, ^{51}Cr, ^{56}Mn, ^{65}Zn, ^{59}Fe, and ^{60}Co. Samples older than 10 days contained 99% ^{65}Zn with traces of ^{51}Cr, ^{59}Fe, and ^{60}Co being present. The ^{65}Zn activity undoubtedly originated from zinc corrosion products dissolving from the admiralty brass condenser and being neutron activated during recycling of the reactor water through the pressure vessel.

The total liquid radwaste discharges for the period May 19, 1966, to October 31, 1967, amounted to 513 mCi. Total gaseous radwaste emissions amounted to 9330 Ci of noble gases, and 259 mCi of particulates. Again, the gaseous discharges were presumed to be mainly from naturally-occurring radionuclides.

Operation of the plant during this period produced only one significant radiation incident. On September 16, 1967, a condenser tube leak occurred which resulted in cross contamination of the cooling tower water with the primary system water. Airborne contamination was also released in the reactor building, although not of a serious magnitude. A total of 69.5 mCi of radioactivity consisting primarily of ^{65}Zn and ^{24}Na was released to the effluent ditch following this incident, but the radioactivity levels in the diluted discharge never exceeded MPC values. Cooling tower blowdown also released an additional 27.9 mCi to the Big Sioux River.

A considerable effort was spent in cleaning up the reactor systems affected by the condenser tube leak. In addition, a major inspection of the reactor revealed the serious deterioration of the steam separators. The reactor fuel, control rods, and vessel internal components were then removed for inspection. Due to the deterioration of the steam separators, the reactor was never restarted, and the plant was converted to a gas/oil fired unit which was used only during periods of peak power demand.

APPENDIX C

PARTIAL DECOMMISSIONING

APPENDIX C

PARTIAL DECOMMISSIONING

The decision was made in August 1968, to terminate the nuclear reactor operations at Pathfinder and install gas/oil-fired boilers to provide steam for the turbine generator. All of the nuclear fuel was removed from the reactor and stored in the storage pool until shipment offsite. Partial decommissioning of the nuclear plant was then commenced for conversion to the fossil-fueled system. Steam, reactor feedwater and other lines penetrating the reactor building were cut, and steel end-caps were welded in place or the open ends were covered with blank flanges. The partial dismantling procedures followed at Pathfinder are summarized as follows.

C.1. REACTOR BUILDING

Dismantling in the reactor building consisted of deactivating the nuclear reactor, removal of external penetrant piping and sealing of the penetrations, draining of piping systems and pools, deactivation of power circuits, and isolation of the building from the rest of the plant.

C.1.1 Nuclear Reactor

The reactor was dismantled by the following actions:

- Control rod drives. The control rod drives were removed, cut in half, and stored in the fuel storage pool under a slab of reinforced concrete. Blind flanges were installed on the control rod drive nozzles and the bolts welded to secure the flanges. The electrical cables to the drives were cut outside the reactor building.
- Reactor vessel. The reactor vessel was filled with gravel for the purposes of shielding and security. The vessel head was bolted in place and the bolts were welded to secure the head. After the head was secured and the reactor system drained, a vacuum was drawn on the vessel to remove most of the remaining water.
- Pool bridges and cranes. The reactor pool bridges were welded in the raised position. The reactor building polar crane and traveling bridge crane power was disconnected and the carriages were welded in fixed positions.

- Air locks. The equipment door and emergency personnel lock in the reactor building were welded closed. The personnel air lock was secured with a special combination lock.
- Other systems. Mechanical penetrations into the reactor building were cut and then sealed by welding caps over all the pipes or penetrations. All equipment and piping systems were drained. The two ventilation ducts were welded shut. A 1/2-in. pressure equalization line containing a filter was routed from the reactor building to the stack. The fire protection system was disconnected.

  All power to the reactor building was disconnected. Lighting was left in a stand-by condition so that it can be reconnected if required for future inspections. All control and instrumentation systems were disconnected.

C.2 FUEL HANDLING BUILDING

C.2.1 The Lower Levels

The fuel handling building below the operating floor serves as the storage area for the activity in the piping and equipment in that building.

- Sealing the area. Modification to the structure included sealing all entrances into the lower levels of the building. Entrances into the lower levels from the operating floor include two stairways and one crane hatch. The stairwells were sealed with concrete. A personnel hatch was left in the north stairway seal to allow personnel to enter the basement area for inspection.

  Penetrations and access ways were sealed to prevent air within the isolated area from entering the turbine building or upper level of the fuel handling building.

  Doors at the mezzanine floor and at the basement floor leading to the turbine building and pipe chases in that area were sealed with concrete.
- Radioactivity control. Activated corrosion products in the fuel handling building basement were contained within piping and equipment systems. Contaminated systems were closed to prevent the spread of radioactive material within the storage area. The major portions of the piping systems were left in their present locations. All wall-penetrant pipes containing activated corrosion products were cut and welded closed inside the fuel

handling building lower levels. Piping was removed from the pipe chases to allow complete chase closure. The building basement itself was sealed as completely as practical. A pressure equalization line containing a filter was routed to the stack.

- Other systems. The radioactive waste disposal system was sealed to prevent the spread of contamination. The ventilation penetrations were welded closed. The electrical systems were disconnected. The lighting system was left in a stand-by condition. Instrumentation systems located in the area were deactivated and the service systems (instrument air, service water, fire, etc.) were disconnected. All equipment and piping systems were completely drained.

  The fuel transfer tube connecting the fuel storage pool and the reactor pool was closed by removing the fuel transfer tube valve and by welding plates on both tube ends. The manhole used for access to the fuel transfer valve was locked closed.

C.2.2 The Upper Levels

The upper levels of the fuel handling building were decontaminated to comply with levels for unmonitored access to qualify them for use as storage areas. All pipes and conduit leading to the fuel handling building lower levels were cut and sealed. Piping and electrical systems that were required in the upper portion of the building were rerouted outside the lower levels of the fuel handling building. A thick reinforced concrete cover was placed over the fuel storage pool. The crane hatch was sealed.

C.3 TURBINE BUILDING

All contaminated material and equipment not being salvaged for use in the fossil system were removed from the turbine building and were stored in the reactor building or the lower levels of the fuel handling building, or were shipped offsite.

The turbine building is being used in the operation of the fossil system and was decontaminated to the extent practical. Phosphoric acid (30%) solution was used to decontaminate the low and higher pressure turbine casing, all turbine parts, the condenser, an inlet steam line, and two of the four feedwater heaters.

The decontamination removed approximately 500 mCi of activity (mostly ^{65}Zn) from the turbine and condenser, 300 mCi from turbine parts, and 500 mCi from the feedwater heaters. A total of 1300 mCi of activity was packaged and shipped offsite. Survey results indicated a decontamination factor of 2 for the turbine and condenser, about 5 for turbine parts, and 10 or greater for the feedwater heaters. In 1969 an estimated 1.5 Ci (mostly ^{65}Zn) remained in the system converted to fossil fuel. Most of this activity remained in place, attached to the piping and metal surfaces, and has undergone radioactive decay. Present radioactivity levels in the boiler water are near background.

Radioactive wastes produced during dismantling activities consisted of contaminated equipment and piping, spent resins, filter cartridges, contaminated liquid wastes, contaminated construction equipment, tools, and debris, plus miscellaneous decontamination materials. Construction debris consisted of equipment and piping small enough to be drummed, piping installation, and other miscellaneous materials. Decontamination materials consisted of paper towels, gloves, plastic bags, etc., used in health physics work.

Equipment and piping too big to be drummed were stored on the lower levels of the reactor building. Every piece of equipment stored there was tagged with an inventory number, identification, and a radiation level reading. A log of the inventory was then kept at the site. Contaminated piping and hardware were also stored in the drained fuel storage basin.

Two hundred eighty-five drums of solids, consisting of construction debris, filter cartridges, and other contaminated materials, and 114 drums of spent resins were shipped offsite for burial by Chem-Nuclear Services, Inc.

Liquid radioactive effluents were at first disposed of by dilution only until it appeared that the proposed annual average discharge guideline of 2×10^{-8} $\mu Ci/m\ell$ might be exceeded. The liquid wastes were then demineralized to the extent practical and then discharged to the drainage ditch at levels below the limits set by the NRC. Between September 1967 and November 1970 a total of 233 mCi of liquid radwastes was discharged to the ditch.

In November 1969, an estimate of the radionuclide inventory at the Pathfinder plant was calculated and is given in Table C.1. Unfortunately, the radionuclide composition was not given. This original inventory has, of course, undergone extensive radioactive decay during the past 12 years. In 1972, another estimated radionuclide inventory was calculated and is broken down as follows:

<u>Reactor Internals (neutron-activated pressure vessel and internals)</u>--15,000 Ci of ^{60}Co; 3,110 Ci of ^{63}Ni; 49,600 Ci of ^{55}Fe

<u>Fuel Handling Building (translocated activated corrosion products in piping and tanks)</u>--1 Ci of ^{60}Co (48%) and ^{65}Zn (52%)

<u>Turbine and Boiler Building After Decontamination (translocated activated corrosion products in piping and metal surfaces</u>--0.25 Ci of ^{60}Co (35%) and ^{65}Zn (65%)

At the time of the present study (September 1980), the ^{65}Zn had decayed to trace amounts and the main radioactivity is due to ^{60}Co, with smaller quantities of long-lived radionuclides such as ^{55}Fe, ^{63}Ni, and ^{108m}Ag.

TABLE C.1. Summary of Radioactivity on Site[a]

Reactor Building	Curies
Boiler Shroud	9250.0
Grid Plate	11.5
Superheater Structure	11000.0
Boiler Boxes	22.0
Steam Dryer	0.13
Steam Separators	550.0
Holddown	330.0
Boiler Control Rods	364.0
Vessel Walls	3.13
Separator Support Shelf	4.5
Feedwater Ring	0.03
Neutron Windows	68.0
Ion Chambers	243.0
Superheater Control Rods	1.3
Pumps and Recirculation Lines	1.02
	21848.61
Fuel Handling Building	
Spent Resin Tank	10.0
Purification Coolers and Pipes	0.12
Flash Tank	0.09
Other Tanks	0.03
Storage Pool	1.0
Other Sources	0.13
	11.37
Turbine and Boiler Building	
Turbine	0.1
Condenser	0.2
Heaters (including deaerator)	0.3
Hydrogen Cooler	0.1
Steam Line (including inlet leads)	0.1
Condensate Pipe	0.1
Boilers	0.01
	0.91

(a) Calculated November 14, 1969 (Neglecting Fuel Elements).

NOTE: Total estimated activity onsite in 1969: 21860.89 Ci.

APPENDIX D

SAMPLE INVENTORY AND DISPOSITION

APPENDIX D

PATHFINDER SAMPLE INVENTORY AND DISPOSITION--SAMPLED JULY 1980

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
1	WP-76 300-A	3 1/2" dia. SS pipe from reactor water purification line--end with 1" pipe welded in at 90°--pipe stored in reactor building	8,500 thru pipe 40,000 at open end of pipe	Cut into following pieces 300A-2 300A-1 300A-4 300A-3
2	WP-76 300-B	Same as No. 1--next 6" long piece of straight pipe	40,000 at open end of pipe	Given to J. R. Divine
3	WP-76 300-C	Same as No. 1--next 6" long piece of straight pipe	45,000 at open end of pipe	Archive sample
4	WP-42 No. 11	1 3/4" dia. hole saw plug from 8" dia. pipe from carbon steel reactor feedwater pump suction--pipe stored in reactor building	<200 outside surface	Directly counted on Ge(Li)--leached for radiochemistry
5	WP-74 301-A	2" dia. hole saw plug from 8" dia. pipe from carbon steel reactor feedwater line--pipe stored in reactor building	30,000 at inner surface of pipe 25 through pipe	Directly counted on Ge(Li)--leached for radiochemistry
6	WP-74 301-B	Same as No. 6	50,000 at inner surface of pipe	Directly counted on Ge(Li)--sent to J. R. Divine
7	Concrete Core No. 1	see accompanying listing		
8	Concrete Core No. 2	see accompanying listing		
9	MSB-A	2" dia. hole saw plug from 6" dia. carbon steel pipe from main steam bypass line sampled at second level of steam chase 5' above grating--reactor building	--	Directly counted on Ge(Li)--leached for radiochemistry
10	MSB-B	Same as No. 9	17,000 at inner surface of pipe	Directly counted on Ge(Li)
11	MSB-C	Same as No. 9	15,000 at inner surface of pipe	Directly counted on Ge(Li)--sent to J. R. Divine

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
12	RFW-A	2" dia. hole saw plug from 8" dia. reactor feedwater line--sampled from 3rd level of steam chase 3' above grating in reactor building	45,000 on inside surface of plug	Directly counted on Ge(Li)--leached for radiochemistry
13	RFW-B	Same as No. 12	55,000 on inside surface of plug	Counted directly on Ge(Li)--sent to J. R. Divine
14	SPC-A	2" dia. hole saw plug from 6" dia. shield pool cleanup line--sampled at Y near hot spot on 3rd level of steam chase in reactor building--heavy corrosion film red on top and yellow on bottom	55,000 on inside surface of plug	Directly counted on Ge(Li)--leached for radiochemistry
15	SPC-B	Same as No. 14	25,000 on inside surface of plug	Directly counted on Ge(Li)--sent to J. R. Divine
16	RLL-A	5" long section of 2 3/8" dia. pipe from reactor liquid level column (lower leg)--from reactor building	100,000; 61 m rad; 1 mR at end of pipe	Not cut up--leached for radiochemistry
17	RLL-B'	Same as No. 16	Same as No. 16	Sent to J. R. Divine
18	TR-A	Strip of SS cut from tool rack on south side of bottom of shield pool--reactor building	100 c/m thru plastic bag	Sent to J. R. Divine
19	TR-B	Same as No. 18	1,900 c/m thru plastic bag	One piece 12" long x 3" wide x 1/4" thick cut into four pieces 3" long--also two 3/4" SS nuts and washers labeled TRB-NW
20	SPD-A	End of ~4" dia. SS drain pipe from bottom of shield pool (NE side of shield pool)--reactor building	1,400 c/m thru plastic bag	
21	SPD-B	Same as No. 20	1,500 c/m thru plastic bag	Sent to J. R. Divine
22	FHR-A	2 1/2" dia. SS pipe section from fuel shoot support strut at bottom of shield pool--reactor building	300 c/m thru plastic bag	14" long piece cut into 3 pieces 4" long piece sent to J. R. Divine 4" long piece for leaching for radiochemistry 4" long piece for archives
23	SPC-A	3 1/2" dia. SS pipe from fuel storage pool cleanup line--sampled at basement of fuel handling building--line from bottom of fuel storage basin to filter--demin. cleanup in FHB basement	80,000	Cut into two pieces 2-3" long for leaching for radiochemistry

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
24	SPC-B	Same as No. 23	80,000	Sent to J. R. Divine
25	PWC-A	2" dia. carbon steel from pool water cleanup pump--discharge to Series 11--pool water for storage basin and shield pool to inlet of No. 11 prefilter--sampled from fuel handling building basement	20,000	Sent to J. R. Divine
26	PWC-B	Same as No. 25	20,000	4" long piece--not cut up--leach as is for radiochemistry
27	PDO-A (no B taken)	2 3/8" dia. SS pipe from inlet to demin. for water from storage and shield pools--sampled from fuel handling building basement	2,000	Cut into three pieces 5" long piece for archive 3 1/2" long piece sent to J. R. Divine 3 1/2" long piece for leaching
28	PDI-A (no B taken)	2" dia. SS pipe from inlet to demin. for storage and shield pools--sampled in SE corner of basement of fuel handling building	15,000	Cut into three pieces 4" long piece for archive 3" long piece sent to J. R. Divine 4" long piece cut into two 2" long pieces for leaching
29	SPCB-A (no B taken)	3 1/2" dia. carbon steel pipe from shield pool coolant bypass-water from shield pool to filter demin.--sampled in SE corner of basement of fuel handling building	2,500	Cut into two pieces 3" long piece sent to J. R. Divine 4" long piece for leaching
30	CWTD-A	2" dia. SS pipe from concentrated waste tank discharge line--sampled from basement of fuel handling building	--	4" long piece--not cut up--leach as is
31	CWTD-B	Same as No. 30	--	Sent to J. R. Divine
32	SRTD	1 3/8" dia. x 7" long SS pipe from spent resin tank discharge line--sampled from basement of fuel handling building	--	Cut in half 3 1/2" long piece sent to J. R. Divine 3 1/2" long piece for leaching
33	HSHT	2" dia. SS elbow pipe from high solids holdup tank--just upstream of suction pump--includes resin and black crud trapped in bend--sampled from basement of fuel handling building	--	Cut into two pieces 3" long straight piece sent to J. R. Divine 4" long elbow for leaching--crud removed from elbow and bagged
34	IFDP	3 1/2" dia. x 12" long SS pipe from inlet line to filter--demin. purification system--sampled from basement of fuel handling building	--	Cut into three pieces 3" long piece sent to J. R. Divine 3" long piece for archives 4" long piece for leaching

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
35	VRM	Brass valve in PVC pipe from high solids manifold line to radiation monitor--sampled from mezzanine level of fuel handling building	>100,000 c/m 45 mRem/hr 25 mR/hr at contact	Opened brass valve and found very coarse gravel-like particles plugging the valve and 1" PVC line--removed particles and bagged--discarded valve and PVC pipe which contained little activity
36	RS-A and B	1 7/8" dia. SS line to reactor sump-pump--sampled at bottom of reactor sump chase--horizontal section before going to filter and pump	1,000 at end of pipe	A. 8" long piece cut into (3) 2 1/2" long pieces for leaching B. 8" long piece sent to J. R. Divine
37	RSE	Black iron elbow connecting 1 7/8" SS pipe to line going to filter and pump of reactor sump--sampled at bottom of reactor sump chase just downstream from RS-A and B	30,000 at end of pipe	Cut off black iron elbow from 4" long piece of SS 1 1/2" dia. pipe--saved pipe and labeled RSE-PIPE
38	WGPT	2 1/2" dia. hole saw plug from 1/2" thick carbon steel waste gas pressurizer tank (large steel tank)--sampled from mezzanine level of fuel handling building where tank stored	<1,000 dpm on inside surface of plug.	Leached for radiochemistry
39	RAVD	8 1/2" x 12" section of 1/16" thick galvanized iron reactor air vent duct--sampled from mezzanine level of fuel handling building where ducts stored	<1,000 dpm on inside surface of duct	Cut up for leaching
40	SHFBS-A	1" dia. SS tube from cluster used for storing superheater fuel elements--sampled from bundle No. 1 in fuel storage basin	3,000	Cut into three pieces 3-4" long for leaching
41	SHFBS-B	Same as No. 40	3,000	Sent to J. R. Divine
42	SHFBS-C	Same as No. 40, except taken from bundle No. 2	15,000	Cut into six pieces 2-3" long for leaching
43	MSL-A	2 5/8" dia. hole saw plug from main steam line--6' long piece stored in fuel storage basin	70,000	Counted directly on Ge(Li) and then leached for radiochemistry
44	MSI -B	Same as No. 43	70,000	Sent to J. R. Divine
45	FRSB-A	Piece of SS fuel rack from fuel storage basin--top piece 2" dia. x 5" long piece from east end of rack	250,000	Cut into two 2" long pieces for leaching

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
46	FRSB-B	Same as No. 45	300,000	Sent to J. R. Divine
47	FRSB-C	Same as No. 45, except cut from west end of rack	45,000	Cut into two 2" long pieces for leaching
48	FRSB-D	Same as No. 47	80,000	Sent to J. R. Divine
49	FTC	Piece of fuel transfer chute--4" x 5" SS piece cut from chute near joining ear--sampled from fuel storage basin	15,000	Cut in half - half sent to J. R. Divine - cut other half in two 2" long pieces for leaching
50	FTTR	Fuel transfer tube roller wheel removed from fuel transfer tube reactor sampled from fuel storage basin	0.5 mR/hr 75 mR/hr	Not cut--leach as is
51	FSBTT	2" dia. x 8" long piece of SS cut from fuel storage basin transfer tube for fuel elements--sampled from fuel storage basin	15,000	Cut in half - half sent to J. R. Divine - other half cut into two 2" long pieces for leaching
52	SCDL	1 3/4" dia. steel line draining main steam line of condensate when reactor was down --colleced from storage drum in cage on basement floor of turbine building--some torch cutting on piece	10,000	Cut off "hot" 4" and then cut that into two 2" long pieces for leaching

REFERENCES

1. Smith, R. I., G. J. Konzek and W. E. Kennedy, Jr. 1978. Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. NUREG/CR-0130, Vol. 1 and 2, prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.

2. Oak, H. D., G. M. Holter, W. E. Kennedy, Jr., and G. J. Konzek. 1980. Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station. NUREG/CR-0672, Vol. 1 and 2, prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.

3. Manion, W. J., and T. S. LaGuardia. 1980. Decommissioning Handbook, DOE/EV/10128-1, prepared for U.S. Department of Energy by Nuclear Energy Services, Incorporated, Automation Industries, Danbury, Connecticut.

4. Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities. 1981. NUREG-0586, U.S. Nuclear Regulatory Commission, Office of Standards Development.

5. Osterhout, M. M., ed. 1980. Decontamination and Decommissioning of Nuclear Facilities. Plenum Press, New York.

6. Witherspoon, J. P. 1981. Technology and Cost of Termination Surveys Associated with Decommissioning of Nuclear Facilities. NUREG/CR-2241. Prepared for U. S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tennessee.

7. Robertson, D. E., et. al. 1981. Transuranium and Other Long-Lived Radionuclides in the Terrestrial Environs of Nuclear Power Plants. EPRI-EA-2045, Project 1059, Final Report. Prepared by Battelle, Pacific Northwest Laboratories for Electric Power Research Institute, Palo Alto, California, September, 1981.

8. USNRC. 1981. Licensing Requirements of Land Disposal of Radioactive Wastes. 10 CFR 61. Federal Register, Vol. 46, No. 142, Friday, July 24, 1981. Proposed Rules.

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Appendix D

Characterization Survey Report for the Pathfinder Plant

CHARACTERIZATION SURVEY REPORT FOR THE PATHFINDER PLANT IN SIOUX FALLS, SOUTH DAKOTA

Revision 0

December 2003

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Attachment 1	Direct Beta Measurement Results
Attachment 2	Removable Alpha and Beta Activity Measurement Results
Attachment 3	Exposure Rate Measurement Results
Attachment 4	Off Site Sample Analysis Results

1.0 Introduction

This report presents the results of the characterization survey performed within the Pathfinder Plant and surrounding environs in the fall of 2003 by Duratek. The characterization survey was performed in accordance with an approved characterization survey plan, Reference 10.1. The results of this characterization survey will be used to help define the scope of future D&D activities at the Pathfinder Plant. It is expected that this report will be forwarded to the NRC as an attachment to the D&D plan currently being prepared. The intent of the D&D plan is to describe those actions necessary to release the Pathfinder Plant and the surrounding environs for free release and terminate its NRC License, License Number 22-08799-02.

2.0 Site Information

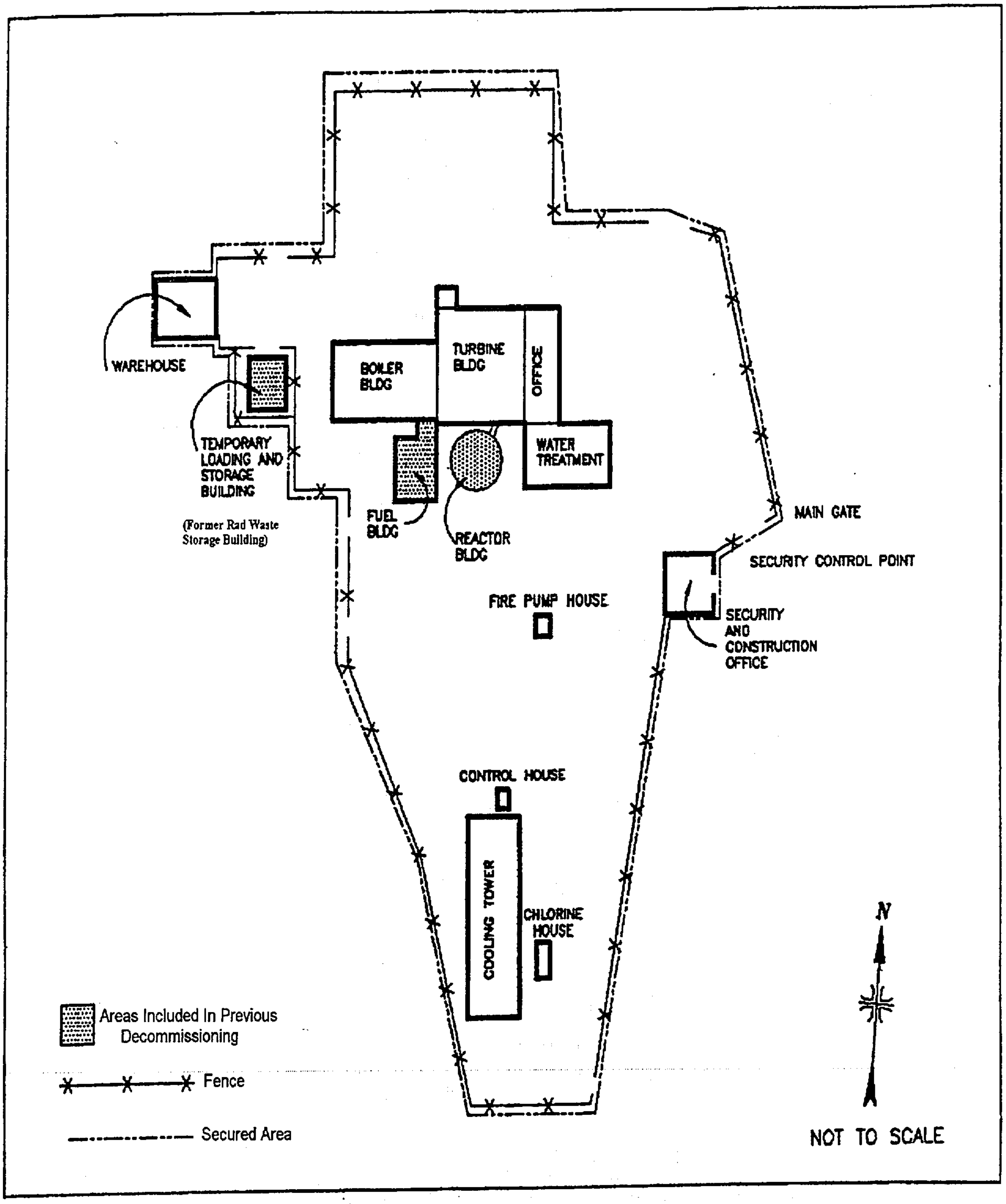
The Pathfinder Plant is located near Sioux Falls, South Dakota in Minnehaha County. The plant is currently owned and managed by Xcel. The Pathfinder Plant was originally designed as an atomic power plant. The Pathfinder reactor achieved initial criticality on March 24, 1964, began commercial operation on August 6, 1966, and ceased operation on September 16,1967 due to a steam separator failure. The nuclear steam supply system (NSSS) was a 66 MWe (203 MWth) boiling water reactor designed by Allis-Chalmers Manufacturing Company of Milwaukee, Wisconsin. The reactor never achieved sustained full power operation. During the final shutdown, a condenser tube leak resulted in the contamination of the service water system and cooling tower basin. These systems were subsequently remediated. The nuclear fuel was shipped off site in 1970 and the plant was placed in SAFSTOR in 1971.

During operation reactor effluents were discharged to the Big Sioux River.

Once operation of the reactor ceased the reactor was isolated from the balance of the plant (BOP) and the BOP was decontaminated in preparation for its conversion into a gas/oil fired peaking station. The reconfigured plant began commercial operation in May 1969. The Pathfinder Plant consists of three package boilers that can burn natural gas and/or number six fuel oil. The steam cycle consists of the boilers, turbine, condenser, low-pressure heaters, deaerator, feed pumps, and high-pressure heater. It continued to operate until July 2000 when the cooling tower collapsed during a storm.

In June 1990 the NRC approved the Decommissioning Plan for the Reactor Building, the Fuel Handling Building, and the Fuel Transfer Vault. Decommissioning of these structures was completed in May 1992 with the issuance of the "Final Survey Report For The Pathfinder Atomic Plant." Following decommissioning the Reactor Building was demolished and the sub basement backfilled and capped. Figure 2.1 shows the current configuration of the Pathfinder Plant.

Figure 2.1
Site Lay Out



The Pathfinder Plant is currently licensed by the NRC, license number 22-08799-02, to possess up to 41 millicuries (mCi) of byproduct materials. This license was issued to account for residual activity remaining in the BOP following the decommissioning of the Reactor Building, the Fuel Handling Building, and the Fuel Transfer Vault.

In February 2003 the NRC was notified that operation of the Pathfinder Plant had been terminated. Decontamination and decommissioning (D&D) activities were initiated with the intent of releasing the site for unrestricted use and termination of license 22-08799-02.

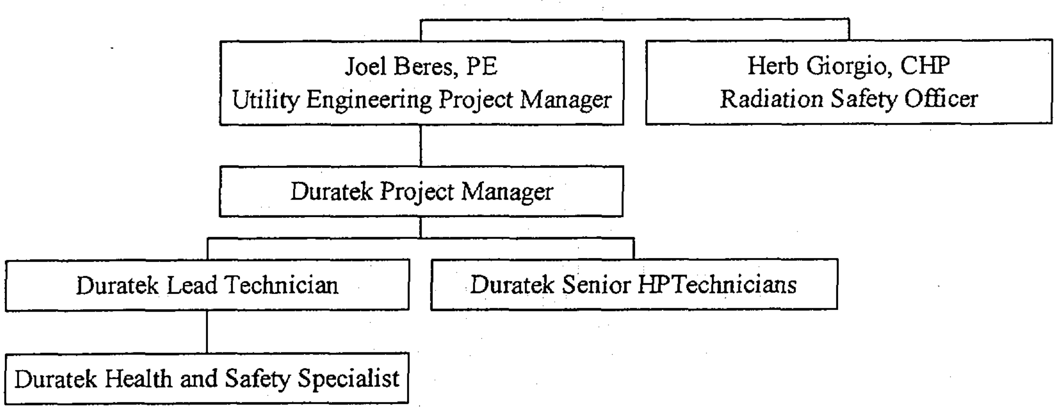
The Angus Anson Plant, which consists of two simple cycle combustion turbines was constructed on the Pathfinder site and placed into operation in September 1994. The Angus Anson Plant and the Pathfinder Plant share a common service water system.

A landfill located behind the plant was used to dispose of non-radioactive waste.

3.0 Organization and Responsibility

The on-site survey team consisted of a project manager, a lead technician, and two senior health physics (HP) technicians and a health and safety specialist. These personnel were experienced in field radiological survey procedures and had current Radiation Worker and HAZWOPER training. An organizational chart is shown in figure 3.0

Figure 3.0
Pathfinder Characterization Survey Team



3.1 Project Manager

The project manager was the primary point of contact and interface with representatives of Utility Engineering who were overseeing the characterization survey for Xcel Energy. He was responsible for the supervision and coordination of daily activities including the technical overview of the characterization surveys. In order to ensure regulatory compliance, the project manager was a certified health physicist (CHP), qualified in the use of the survey instrumentation, familiar with the isotopes of concern, and experienced in performing surveys in accordance with the guidance contained in NUREG-1575 and the characterization survey plan.

3.2 Lead Technician

The lead technician was responsible for health and safety and coordinating activities associated with the characterization survey. He was qualified in the use of the survey instruments and the performance of surveys in accordance with NUREG-1575 and the characterization survey plan. The lead technician was certified in first aid and CPR.

3.3 Health Physics (HP) Technician

The HP technician(s) were responsible for performing the characterization surveys, collecting samples, downloading survey data, and analyzing sample data as necessary. They were qualified in the use of the survey instruments and the performance of surveys in accordance with NUREG-1575 and the characterization survey plan.

3.4 Health and Safety Specialist

The Health and Safety Specialist was responsible for coordinating all confined space entries performed during the performance of the characterization survey. He was familiar with the requirements of 29 CFR 19.10 and the Xcel and Duratek procedures dealing with confined space entries. The Health and Safety Specialist was on site during all confined space entries

3.5 Off Site Project Support Personnel

Instrumentation support personnel supported the characterization survey by providing calibrated instrumentation and certified check sources. A Health and Safety Specialist addressed issues related to legionaries disease during the survey of the cooling tower basin and the condenser. Certified health physicists and senior radiological engineers assisted with the preparation of the characterization survey plan and this characterization survey report.

3.6 Subcontractors

Eberline Services located in Oak Ridge, TN performed all of the off site analyses on the samples collected as part of the characterization survey.

4.0 Survey Overview

The areas surveyed in and around the Pathfinder Plant during the characterization survey include:

- Accessible building surfaces in the basement of the Turbine Building. The primary focus was on the floor, walls, and horizontal surfaces in overhead spaces.
- Systems and structures located in the basement of the Turbine Building including sumps, drain troughs, area beneath the condenser, condenser hot well, condensate system, feed water pump, and feed water heater.
- Accessible building surfaces on the mezzanine in the Turbine Building. The primary focus was on the floor, walls, and horizontal surfaces.
- Systems and structures located on the mezzanine in the Turbine Building including hydrogen coolers, drain cooler, air ejector, and steam supply (steam trap drain, main steam line, stop and control valves).
- Accessible building surfaces on the turbine deck. The primary focus was on the floor, walls below 2 meters, and horizontal surfaces in overhead spaces.
- Systems and structures located on the turbine deck including high pressure turbine blades, HVAC, and floor plugs.
- Accessible building surfaces on the first floor of the Boiler Building. The primary focus was on the floor and walls below 2 meters.
- Systems and structures located on the first floor of the Boiler Building including the mud drums, drain troughs, floor drains, sump, and flash tank.
- Accessible building surfaces on the second floor of the Boiler Building. The primary focus was on the floor and horizontal surface.
- Systems and structures located on the second floor of the Boiler Building including the de-aerator, exhaust fan, and stack.
- Accessible building surfaces within the Warehouse and the Security Building. The primary focus was on the floors.
- Ground water wells. The primary focus was on collecting water samples.
- Settling basins. The primary focus was on collecting sediment samples.
- Effluent discharge pathways. The primary focus was on collecting surface, (0 to 6 inch), and sub surface, (6 to 12 inch), soil/sediment samples.
- Catch basins and sumps surrounding the plant.
- Cooling Tower Basin. The primary focus was on the floor and collecting sediment samples.

To facilitate the characterization survey Xcel crews opened, disassembled, or cut access ports in many of the systems to be surveyed to allow access to the system's internals.

4.1 Data Quality Objectives

The data quality objectives for the characterization survey included the following:

- Collecting sufficient data to accurately identify contaminated areas and systems.
- Collecting sufficient samples to accurately identify the radionuclides of interest.
- Collecting sufficient data to accurately determine the relative fractions of the radionuclides of interest.
- Collecting sufficient data to estimate the scope of future decommissioning activities including estimated radioactive waste volumes

4.2 Radionuclides of Concern

The 1982 Topical Report, Residual Radionuclide Distribution and Inventory at the Pathfinder Generating Plant, reference 10.4, provides information on the radionuclides that were present at the Pathfinder plant. Since the reactor ceased operation in 1967 mainly those radionuclides with a half-life exceeding 5 years (approximately 7 half lives) have a potential of being present in significant quantities. Fe-55 (half-life of 2.7 yrs) is also included due to the high activity concentrations in activated concrete for the Pathfinder plant noted in NUREG/CR-3474, *Long Lived Activation Products in Reactor Materials*, reference 10.5. Ag-108m has been included because it was seen in several piping systems including the main steam (reference 10.4, June 1982). Table 4.2 lists some of the potential radionuclides that were considered during the characterization survey.

Table 4.2
Potential Radionuclides of Concern

Radionuclide	Half Life	Major Radiations, Energies and Intensities		
		Alpha	Beta (average)	Gamma
H-3	1.228E1 yrs		5.685 keV 100%	
C-14	5.730E3 yrs		49.47 keV 100%	
Fe-55	2.7E0 yrs			Low energy x-rays
Ni-59	7.5E4 yrs			Low energy x-rays
Co-60	5.27E0 yrs		95.79 keV 100%	1173 keV 100% 1332 keV 100%
Ni-63	1.001E2 yrs		17.13 keV 100%	
Sr-90/Y-90	2.86E1 yrs		195.8 keV 100% 934.8 keV 100%	
Nb-94	2.03E4 yrs		145.8 keV 100%	702.6 keV 100% 871.1 keV 100%
Tc-99	2.13E5 yrs		84.6 keV 100%	
Ag-108m	1.27E2 yrs			433.9 keV 89.9% 614.4 keV 90.4%

Radionuclide	Half Life	Major Radiations, Energies and Intensities		
		Alpha	Beta (average)	Gamma
				722.9 keV 90.5%
I-129	1.57E7 yrs		40.9 keV 100%	39.58 keV 7.5%
Ba-133	1.05E1 yrs			356 keV 60 %
Cs-137	3.017E1 yrs		156.8 keV 94.6% 415.2 keV 5.4%	
Eu-152	1.36E1 yrs		300.8 keV 27.8%	121.8 keV 28.4% 964 keV 14.4% 1085.8 keV 10% 1112 keV 13.3% 1407 keV 20.7% 344 keV 26.5% 778.9 keV 12.7%
Eu-154	8.8E0 yrs		225.4 keV	123.1 keV 40% 1274 keV 35.5%
Eu-155	4.96E0 yrs		45.2 keV 100%	86.5 keV 31% 105.3 keV 20.7%
Pb-210	2.226E1 yrs		4.14 keV 80.2% 16.13 keV 19.8%	46.5 keV 4.05%
Th-230	7.7E4 yrs	4670 keV 100%		67.67 keV .373%
Th-232	1.405E10 yrs	3979 keV 100%		59.0 keV .19%
U-234	2.45E5 yrs	4724 keV 27.4% 4776 keV 72.4%		
U-235	7.0E8 yrs	4396 keV 55%		143.8 keV 10.5% 185.7 keV 54%
U-238*	4.51E9 yrs	5150 keV 25% 4200 keV 75%	103 keV 21% 193 keV 79%	63 keV 3.5% 93 keV 5.4%
Pu-238	8.775E1 yrs	5457 keV 28% 5499 keV 72%		
Pu-239	2.41E4 yrs	5104 keV 11.5% 5142 keV 15.1% 5155 keV 73.3%		
Pu-241	1.44E1 yrs		5.23 keV 100%	
Am-241	4.32E2 yrs	5443 keV 13% 5486 keV 85%		59.5 keV 35.9%

* The beta particles and gamma photons for U-238 are from the Th-234 daughter.

4.3 Instrumentation

The selected survey instrumentation and count times ensured that their sensitivities were sufficient to detect the identified radionuclides at the minimum detection requirements. Table 4.3 provides a list of the instruments used during the characterization survey.

The Ludlum Model 2350 Data Logger was used in combination with a large area gas flow proportional detector for obtaining measurements of total beta activity and for performing beta scans. The Data Logger is a portable microprocessor computer based counting instrument. The Data Logger is designed to operate with a wide variety of detectors. It was used in combination with sodium iodide detectors for obtaining exposure rate measurements and with a cylindrical gas flow proportional detector for obtaining total beta activity measurements within pipes, drains, etc.

Analysis for removable alpha and beta activity was performed using an Eberline SAC-4 and BC-4 respectively.

Alpha instrumentation was calibrated using Th-230 sources and beta instrumentation was calibrated using Tc-99 sources. The major radiations, including energies and intensities, for both Tc-99 and Th-230 are provided in Table 4.2. The use of instrumentation calibrated to Tc-99 and Th-230 is considered conservative.

Table 4.3
Survey Instrumentation

Instrument/Detector	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum Model 2350 with. 43-68 detector	Gas-Flow proportional ($126cm^2$)	Alpha Beta	^{230}Th ^{99}Tc	Total Alpha and Beta Measurement and Beta Scans.
Eberline BC-4	Shielded GM	Beta	^{99}Tc	Smear Counting
Eberline SAC-4	Zinc Scintillator	Alpha	^{230}Th	Smear Counting
Ludlum Model 2350 with 44-2 detector	NaI (Tl) Scintillator	Gamma	^{137}Cs	Exposure Rates
Ludlum Model 2350 with PSL 3R detector	Gas-Flow Proportional Pipe Detector	Beta	^{99}Tc	Total Beta Activity Measurements

4.4 Calibration

The Data Loggers, associated detectors and all additional portable instrumentation are calibrated on a semi-annual basis using National Institute of Technology, NIST, traceable sources and calibration equipment. Procedures for calibration, maintenance, accountability, operation, and quality control of instrumentation implement the appropriate guidance established in American National Standard Institute, ANSI, standards ANSI N323-1978 and ANSI N42.17A-1989.

Instrument calibration included:

- High voltage calibration
- Discriminator/threshold calibration
- Window Calibration
- Alarm operation verification
- Scaler calibration verification

Detector calibration included:

- Operating Voltage determinations
- Calibration constant determinations
- Dead time correction determinations

Calibration labels showing the instrument identification number, calibration date, and calibration due date were attached to all instruments. Control charts and/or source check criteria were established prior to the initial use of the instrument. All instrumentation was inspected and source checked daily, before use, to verify calibration status and proper operation.

4.5 Sources

All sources used for calibration or efficiency determinations were representative of the instrument's response to the expected radionuclides and were traceable to NIST.

Health physics technicians controlled all radioactive sources used for instrument response checks and efficiency determination. Sources were stored securely when not in use.

4.6 Off Site Sample Analysis

Eberline Services analyzed all samples collected during the characterization survey, with the exception of smears collected for accessing removable alpha and beta activity. All samples were analyzed by gamma spectroscopy in order to identify and quantify gamma emitting radionuclides that may be present. Liquid samples were also analyzed for H-3. Select samples, primarily those suspected of containing licensed radioactive material, were also analyzed for H-3, C-14, Fe-55, Ni-59, Ni-63, Sr-90, Tc-99, Pu-238, Pu-239/240, Pu-241, isotopic thorium and isotopic uranium.

5.0 Minimum Detectable Activity

Minimum Detectable Activity (MDA) is defined as the smallest amount or concentration of radioactive material that will yield a net positive count with a 5% probability of falsely interpreting background responses as true activity. The MDA is dependent upon count times, geometry, sample size, detector efficiency, background, and for the scanning rate and the efficiency of the surveyor.

The MDAs for the direct beta measurements and for the analyses of removable beta activity, was set less than 1,000 dpm/100 cm^2 and 200 dpm/100 cm^2 respectively. The MDAs for analyses of removable alpha activity was set less than 20 dpm/100 cm^2.

Beta scans were performed by positioning the detector a half inch or less from the surface being scanned and scanning at a rate not to exceed 1 detector width per second. Monitoring the audible output of the survey meter resulted in an MDA in the range of 4,000 dpm/100 cm^2 to 5,000 dpm/100 cm^2.

Direct Measurements

The equation used for calculating the MDA for direct measurements is:

$$MDA = \frac{\frac{2.71}{t_s} + 3.29\sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{E\left(\frac{A}{100}\right)}$$

Where:

Where:	MDA	=	Minimum Detectable Activity (dpm/100 cm^2)
	R_b	=	Background Count Rate (cpm)
	t_b	=	Background Count Time (min)
	t_s	=	Sample Count Time (min)
	A	=	Detector Area (cm^2)
	E	=	Detector Efficiency (c/d)

Beta Scans

The equation used for calculating the MDA for beta scans is:

$$MDA = \frac{d' * \sqrt{b_i} * \frac{60}{i}}{E_i * E_s * \sqrt{p} * \frac{A}{100}}$$

Where:	MDA	= Minimum Detectable Activity (dpm/100 cm^2)
	d'	= Decision error taken from Table 6-5 of MARSSIM
	i	= Observation counting interval (scan speed divided by the detector width)
	b_i	= Background count per observation interval
	E_i	= Detector Efficiency (c/d)
	E_s	= Surface Efficiency (typically around 50% for beta contamination on concrete)
	p	= Surveyor Efficiency (typically 50%)
	A	= Detector Area (cm^2)

MDAs for field measurements are shown in table 5.1.

Table 5.1
Minimum Detectable Activities for Field Measurements

Measurement		MDA
Smears	Gross Alpha Gross Beta	< 20 dpm/100 cm^2 alpha <200 dpm/100 cm^2 beta
Direct Beta Fixed Point	Total Fixed Beta Activity	< 1000 dpm/100 cm^2
Beta Scans	Total Fixed Beta Activity	< 5000 dpm/100 cm^2

Table 5.2 summarizes the radionuclide specific MDAs required of the off site laboratory supporting the characterization surveys.

Table 5.2
Minimum Detectable Activities for Radiochemical Analysis

Radionuclide	MDA
Co-60, Cs-137, Ag-108m	0.1 pCi/g
H-3	10 pCi/g
C-14	5 pCi/g
Fe-55	10 pCi/g
Ni-59	20 pCi/g
Ni-63	10 pCi/g
Sr-90	2 pCi/g
Tc-99	10 pCi/g
Pu-241	1 pCi/g
Plutonium-238, 239/240	1 pCi/g
Isotopic Thorium	1 pCi/g
Isotopic Uranium	1 pCi/g

6.0 Survey Design and Implementation

The purpose of the characterization survey was to collect sufficient survey data to characterize Pathfinder BOP areas. The project technician(s) performed the survey in accordance with the guidance and requirements contained in specific survey packages, implementing procedures and the characterization survey plan. The survey design included the following.

- Survey instrumentation was set up and source checked to ensure proper operation.
- The survey team performed preliminary inspections of the areas to identify specific survey requirements and safety concerns.
- The project manager and/or the lead technician prepared specific survey packages for each area to be surveyed.
- The technician performed measurements using calibrated instruments. Daily source and background checks were performed prior to and following each day's measurements.
- Direct survey data was downloaded from the survey instrument into a database for storage and processing. Downloaded files were printed and stored in the appropriate survey package along with any other appropriate information.
- The project manager reviewed the completed survey packages to ensure that all required surveys were performed.
- The project manager reviewed the survey results to identify any areas exceeding the specified release criteria.

6.1 Survey Unit Classification

Survey units are discrete areas, consisting of building surfaces, of a specific size and shape. Survey units are defined based on the site's history, potential for residual contamination, and physical characteristics. All impacted areas are divided into survey units. Impacted areas are those areas with a potential of being contaminated. Non impacted areas are areas that do not have a potential for being contaminated and are not surveyed as part of the characterization survey. Survey units are classified as Class1, 2, or 3. A Class 1 survey unit is a survey unit that has or had prior to remediation contamination levels approaching or exceeding the preliminary derived concentration guideline level, (DCGL) of 5000 dpm/100 cm^2. A Class 2 survey unit is a survey unit that does not have contamination approaching or exceeding the preliminary DCGL. Typically Class 2 survey units are not remediated. If remediation is required the survey unit is reclassified as Class 1. A Class 3 survey unit is a survey unit that is not expected to contain contamination or is expected to contain contamination at a fraction (< 10 %) of the preliminary DCGL.

A preliminary DCGL of 5000 dpm/100 cm^2 (total beta activity) was established to guide the characterization survey. The actual DCGL used during the D&D of the Pahtfinder Plant will be based on licensee commitments, the radionuclides present, and their relative fractions. The preliminary DCGL is assumed to be conservative.

Survey units are limited in size to ensure adequate survey coverage. The type of measurements and scanning frequency is also defined for each survey unit classification. The size limit and scanning frequency for the different classifications of survey units are provided in Table 6.1

Table 6.1
Survey Units

Impacted Survey Unit Classification	Size Limit, m^2	Scan Frequency
Class 1 Building Surfaces	< 100	100 % coverage
Class 2 Building Surfaces/ Structures	<1,000	10 – 100% coverage
Class 3 Building Surfaces/ Structures	No limit	Judgment

During the characterization survey all survey units were classified as Class 3.

6.2 Survey Package Development

For each survey area/unit, the project team developed a survey package, by performing a walk-down and preparing a worksheet/tracking sheet outlining the general survey instructions, location codes, and any specific survey instructions for any abnormal conditions within the survey area. Completion and review signature blocks were used to track the progress of the surveys.

During the survey, the technician updated the survey package(s) with the survey data and results of any special surveys or sample analyses performed.

A total of 12 survey packages were prepared to facilitate the characterization survey. Table 6.2 provides a list of the survey packages. It should be noted that survey packages 10 and 12 were not used.

Table 6.2
Survey Packages

Package 3	Classification	Description
01	Class 3	Overhead in the Turbine Building
02	Class 3	Turbine Deck
03	Class 3	Outside Structures
04	Class 3	Effluent Discharge Pathway
05	Class 3	First Floor of Boiler Building
06	Class 3	Second Floor of Boiler Building
07	Class 3	Mezzanine Of Turbine Building
08	Class 3	Basement Of Turbine Building
09	Class 3	Rad Waste Storage Room (Turbine Building Mezzanine)
10	NA	Not Used
11	Class 3	Water Treatment Room
12	NA	Not Used
13	Class 3	Maintenance Shop (Southwest Side Of Building)
14	Class 3	Fuel Storage Building

6.3 Survey Protocols/Requirements

The characterization survey consisted of beta scans, fixed beta measurements, fixed gamma measurements (in some locations) smears for gross alpha and beta analysis, and samples for off site analysis. Surveys were performed as follows:

6.3.1 Beta Scans

For the class 3 survey units, beta scans were performed over approximately 10% of the accessible building surfaces. Beta scan speeds were established such that contamination at levels exceeding approximately 80% of the preliminary DCGL (5000 dpm/100 cm^2 beta) should have been detected. The scans were performed by positioning the detector approximately a half inch above the surface to be scanned and moving the detector at a rate of less than one detector width per second and monitoring the audible output of the detector.

6.3.2 Direct Beta Measurements

Direct beta measurements were taken on the structural surfaces of the Pathfinder BOP within each survey area/unit. A minimum of 30 measurement locations were collected in each survey unit.

6.3.3 Removable Alpha and Beta Activity Measurements

Smears were collected and analyzed for removable alpha and beta activity throughout the plant and within five sumps surrounding the plant. See Figure 6.3.3 for the locations of the sumps from which smears were collected.

6.3.4 Biased Samples

Biased samples were collected for off site analysis at numerous locations throughout the plant. The sampling locations were divided amongst four buildings and/or structures. Sampling locations included;

- The Cooling Tower Basin. Two samples were collected from within the Cooling Tower Basin.

- The Fuel Building. Two samples were collected from within the Fuel Building. One from within a drain and another from within a shallow sump.

- The Boiler Building. Two samples were collected from within the Boiler Building. One from within a mud drum and another from within the de aerator tank.

- The Turbine Building. Seven samples were collected from within the Turbine Building. The sample locations within the Turbine Building included:

    1. The Hydrogen Cooler on the Turbine Building Mezzanine

    2. The condenser cold side

    3. The condenser hot side

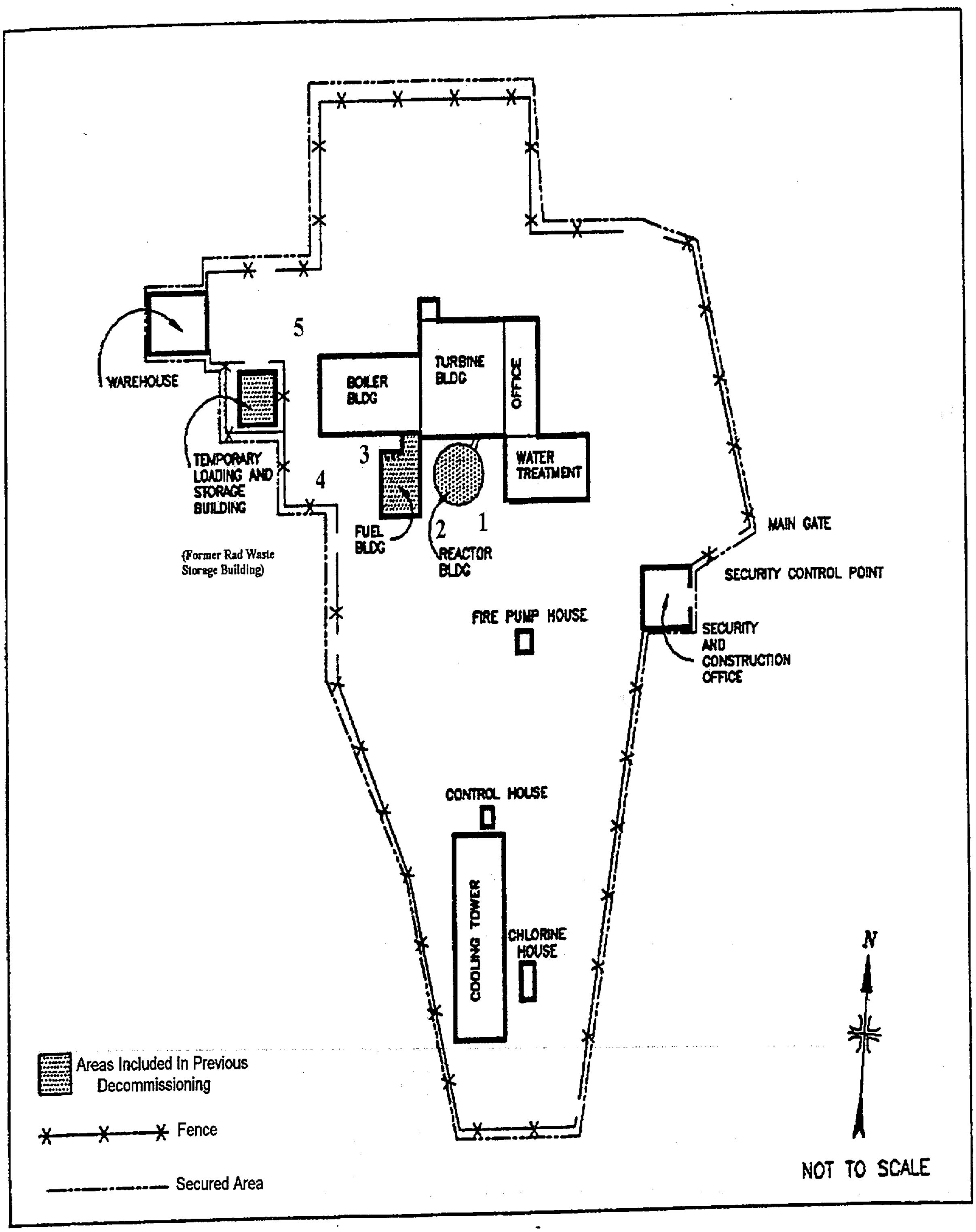
    4. The condensate pit in the basement of the Turbine Building

    5. The condenser expansion joint

    6. The Turbine Building hot side sump

    7. The Turbine Building cold side sump

Figure 6.3.3
Sump Locations



6.3.5 Exposure Rate Measurements

Exposure rate measurements were taken approximately 1 meter above the floor in both the Utility Building and the former Rad Waste Storage Building due to the amount of material being stored within the buildings at the time of the characterization survey.

6.3.6 Environmental Samples

Environmental samples were collected for off site analysis at numerous locations surrounding the plant. Sampling locations included;

- Nine ground water wells labeled Well P1 through P9. See Figure 6.3.6.1 for the well locations.

- Seven locations in the effluent discharge pathway labeled Location 1 through 7. When sampling conditions allowed both surface (0 – 6 inches), and subsurface (6 – 12 inches) samples were collected. See Figure 6.3.6.2 for the sample locations in the effluent discharge pathway.

- Eight locations in the four settling basins, two locations in each of the basins. Surface (0 – 6 inches), and subsurface (6 – 12 inches) samples were collected at each of the sampling locations. See Figure 6.3.6.3 for the sample locations within the settling basins

- Three locations at the base of the cooling tower basin, adjacent to the paved area. Surface (0 – 6 inches), and subsurface (6 – 12 inches) samples were collected at each of the sampling locations. See Figure 6.3.6.4 for the sample locations at the base of the cooling tower.

Figure 6.3.6.1
Ground Water Well Locations

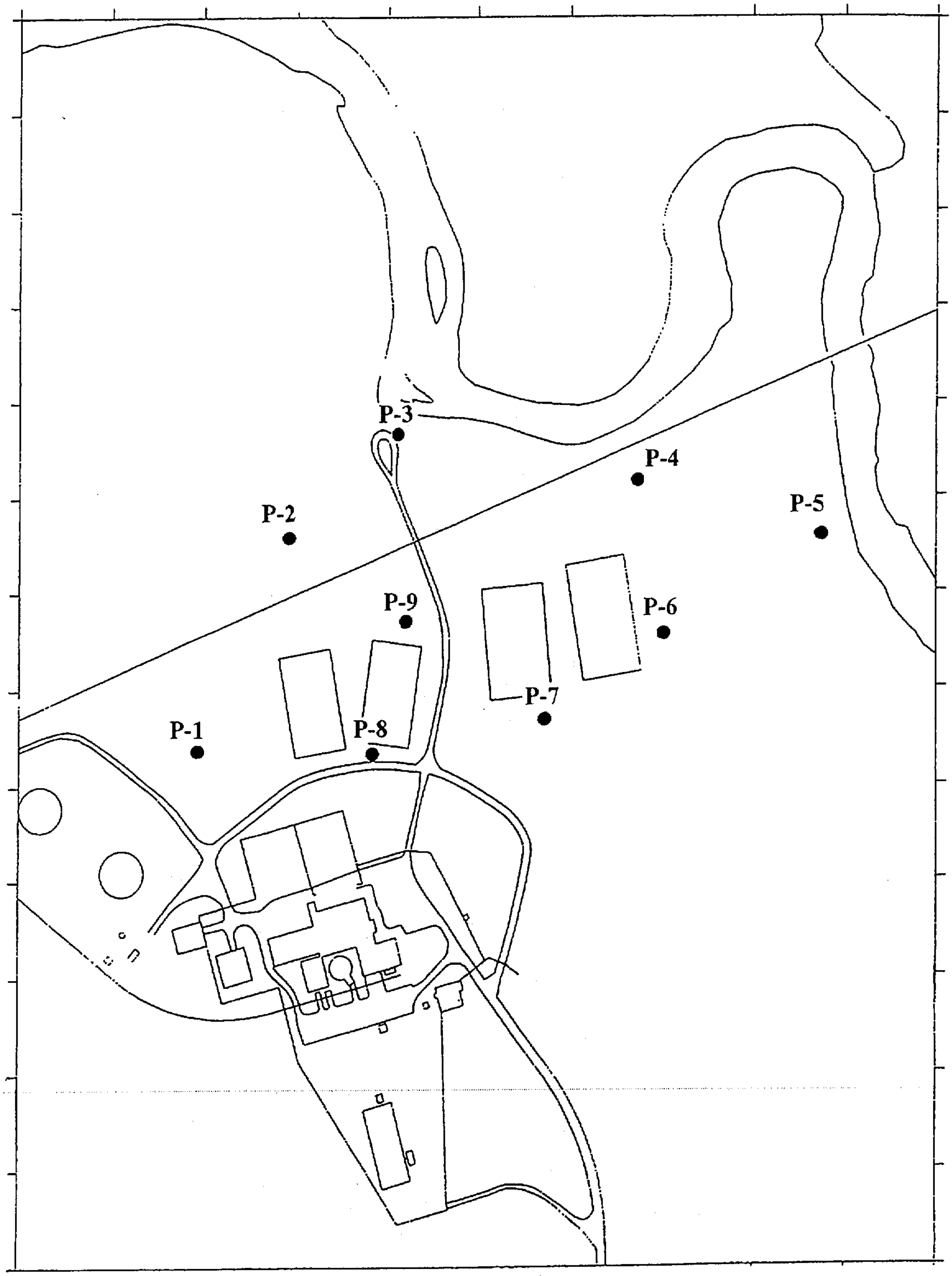


Figure 6.3.6.2
Sampling Locations In Effluent Discharge Pathway

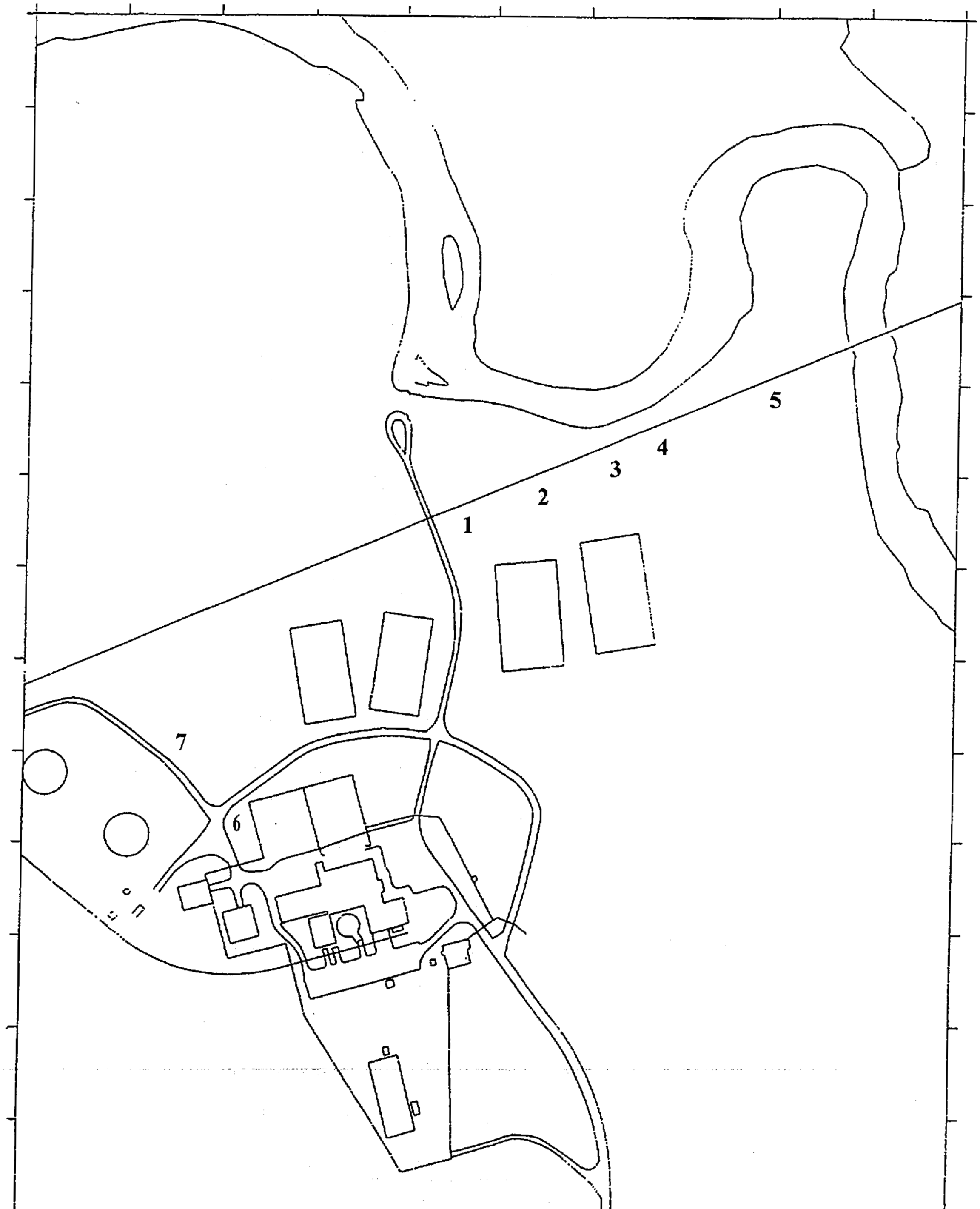


Figure 6.3.6.3
Sampling Locations In Settling Basin

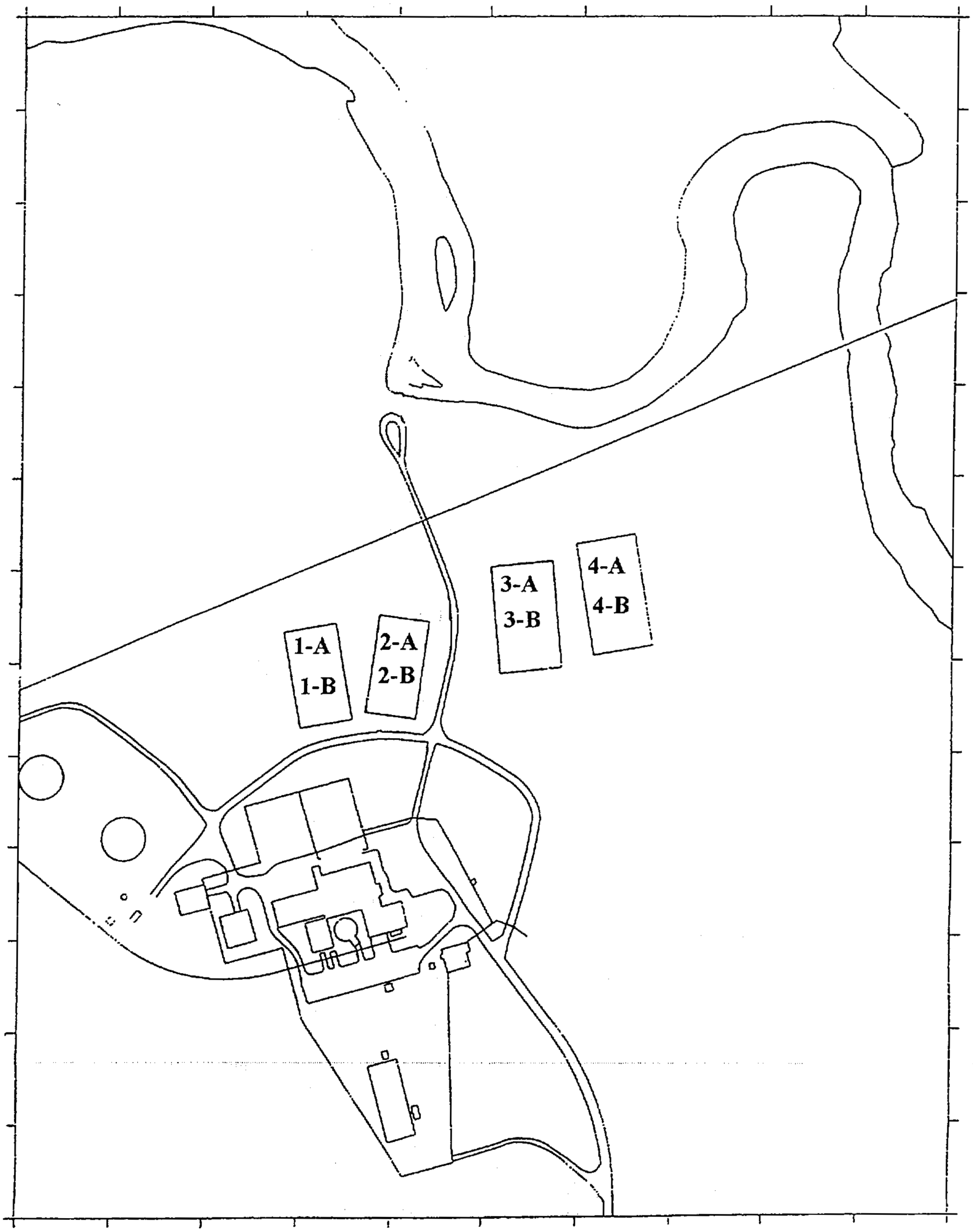
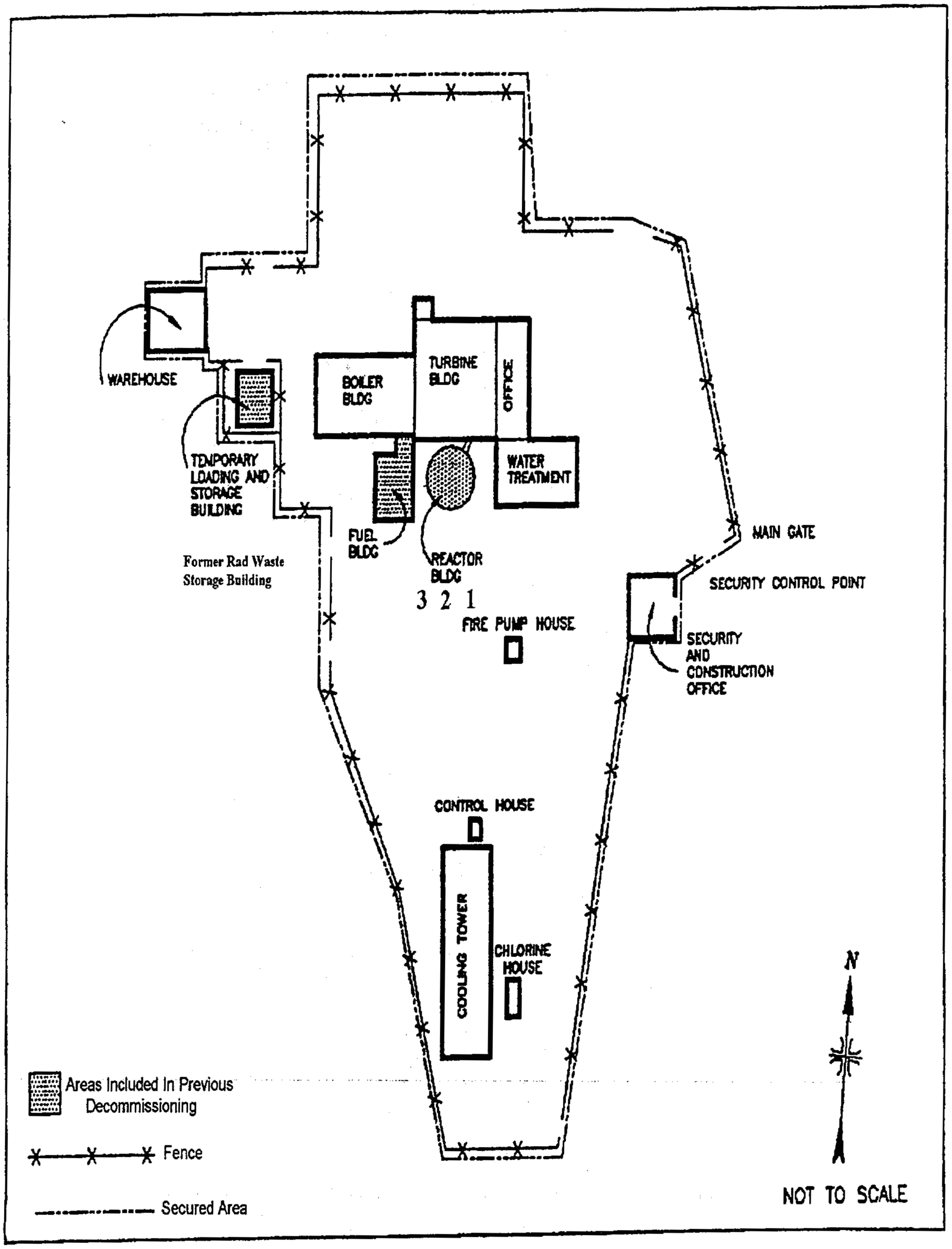


Figure 6.3.6.4
Sampling Locations At The Base of Cooling Tower

7.0



7.0 Quality Assurance and Quality Control

Once the surveys were complete, the data was assessed and evaluated to ensure that all the requirements of the characterization survey plan were met and that the data was acceptable.

Duratek's Quality Assurance/Quality Control Programs ensured that quality and regulatory requirements were satisfied. All activities affecting quality were controlled by procedures or the characterization survey plan. These documents included the following Quality Control measures as an integral part of the survey process.

Selection of Personnel

Project management and supervisory personnel were required to have extensive experience with Duratek procedures and to be familiar with the requirements of MARSSIM and the characterization survey plan. Management had prior experience with the radionuclide(s) of concern and a working knowledge of the instruments used to detect the radionuclides on site. Project management and supervision were required to maintain OSHA safety qualifications as safety is a primary concern of Duratek.

Duratek selected supervisory personnel to direct the survey based upon their experience and familiarity with the survey procedures and processes. Likewise, Health Physics technicians who performed the surveys were selected based upon their qualifications and experience.

Training

All project personnel received site-specific training to identify the specific hazards present in the work and survey areas. Training included a review of the characterization survey plan and Duratek procedures. During site orientation and training, survey personnel become familiar with site emergency procedures.

Written Procedures

All survey tasks, which are essential to survey data quality, were controlled by procedures and the characterizations survey plan.

Instrumentation Selection, Calibration and Operation

Duratek selected instruments proven to reliably detect the radionuclides present at the facility. Duratek or qualified vendors calibrated instruments under approved procedures using calibration sources traceable to the National Institute of Standards and Technology (NIST).

All instruments and detectors were inspected and source checked daily when in use to verify proper operation. Control charts and/or source check criteria were established at the beginning of the project for reference.

Procedures for calibration, maintenance, accountability, operation and quality control of radiation detection instruments implement the guidelines established in American National Standard Institute (ANSI) standard ANSI N323-1978 and ANSI N42.17A-1989.

Survey Documentation

Survey packages were the primary method of controlling and tracking the hard copy records of survey results. Records of surveys were documented and maintained in the survey package for each area according to Duratek procedures. Each survey measurement was identified by the date, technician, instrument type and serial number, detector type and serial number, location code, type of measurement, mode of instrument operation, and Quality Control (QC) sample number, as applicable.

Chain of Custody

Procedures established responsibility for the custody of samples from the time of collection until results are obtained. Samples shipped off site for analysis were accompanied by a chain-of-custody record to track each sample.

Records Management

Generation, handling and storage of survey data packages were controlled by an approved procedure.

Duplicate Review of Survey Results

The survey package and survey data from each area was reviewed by two separate people to verify all documentation was complete and accurate. This included the surveyor and either the project manager or his designee.

8.0 Survey Results

8.1 Beta Scans

Beta scans were performed on approximately 10% of accessible building surfaces. No areas with contamination in excess of the preliminary DCGL was identified during the scanning process.

8.2 Direct Beta Measurements

The results of the direct beta measurements taken as part of the characterization survey are summarized in Table 8.2. Attachment 1 contains the spreadsheets of the individual direct beta measurements. The direct beta measurements results have not been corrected for natural radioactivity present in many materials of construction.

Table 8.2
Direct Beta Measurement Summary

Package #	Surface or Structure	Num of Meas	Mean	Max	Standard Deviation	MDA
			dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
1	Overhead in the Turbine Building	30	127	399	137	297
1	Crane Catwalk	10	145	284	93	297
2	Floor Plugs	40	-91	446	244	290
2	Internals of High Pressure Turbine	20	456	1,455	441	267
2	Floor	30	162	376	80	278
2	Walls Below 2 Meters	43	242	1,407	493	278
2	Ventilation Duct Internals	26	4,186	**18,915**	6444	296
2	Ventilation Duct Internals (re-count on filters)	6	1,745	1,992	195	306
3	Cooling Tower Basin Floor	15	240	368	101	290
3	Security Building Floor	30	302	477	96	290
3	Rad Waste Storage Building Floor	30	343	552	98	290
3	Utility Building Floor	30	412	627	122	290
4	Sump 1	10	270	356	95	287
4	Sump2	10	431	716	216	287
4	Sump 3	10	328	641	178	287
4	Sump 4	10	480	607	104	287
4	Sump 5	6	620	811	120	287
5	Floor Drains	8	310	516	168	240
5	Mud Drum (right side)	10	325	1,911	567	240
5	Mud Drum (left side)	10	71	304	98	240
5	Inside Boiler Tank (back side)	10	958	1,577	560	287
5	Flash Tank	14	32	463	321	287
5	Sump	10	-131	3	97	287
5	Horizontal Surfaces	30	-25	435	163	287
5	Floor	30	261	456	100	287
5	Drain Troughs	11	390	540	121	287

Package #	Surface or Structure	Num of Meas	Mean	Max	Standard Deviation	MDA
			dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
6	Horizontal Surfaces	30	199	478	133	251
6	De-aerator Tank	10	-153	96	116	251
6	Exhaust Fan	10	-117	30	97	251
6	Floor	30	427	609	94	251
6	Stack	10	30	236	185	250
7	Horizontal Surfaces	45	158	823	181	236
7	Main Steam Line	15	116	289	106	236
7	Air Ejectors	10	332	1,812	543	236
7	Walls	45	357	1,281	369	229
7	Steam Drain Trap	10	-61	178	130	229
7	Floor	45	313	669	188	229
7	Stop Valve	5	1,289	2,174	535	257
7	Control Valve	10	1,153	3,020	1,117	257
7	Hydrogen Cooler	10	-13	271	163	245
7	Floor Of Electrical Room	20	301	480	111	291
7	Floor Of Instrument Shop	20	320	705	130	291
7	Floor Of Room East Of Rad Waste Room	20	290	480	123	291
7	Floor Of Room South Of Rad Waste Room	20	66	290	152	291
8	Sub Surface Drain Up Stream Of Condenser	16	1,999	**8,143**	2,397	653
8	Sub Surface Drain Down Stream Of Condenser	27	10,253	**38,714**	7,233	653
8	Sub Surface Drain #1 To Hot Side Sump	1	1,405	1,405	NA	653
8	Subsurface Drain #2 To Hot Side Sump	1	429	429	NA	653
8	Condensate System	15	1,292	**12,631**	3174	245
8	Walls	31	250	597	166	245
8	Feed Water Pump	15	-4	178	125	229
8	Feed Water Heater	15	63	223	84	229
8	Floor Under Condenser	20	969	**8,075**	1731	242
8	Floor	30	277	470	121	242

Package #	Surface or Structure	Num of Meas	Mean	Max	Standard Deviation	MDA
			dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
8	Horizontal Surfaces	30	-38	218	91	242
8	Condenser Hot Well	20	1,518	**5,849**	1,869	259
8	Upper Level Of Condenser	20	166	466	199	259
8	Condenser Expansion Joint	30	1,872	**12,757**	3,662	239
8	Hot Side Sump	16	348	603	146	239
8	Trenches	30	370	836	216	239
8	Floor Of Maintenance Shop	30	273	739	144	272
8	Floor Of Maintenance Office	20	365	562	119	272
8	Floor Of Laundry	20	326	582	126	272
9	Horizontal Surfaces	20	77	528	145	269
9	Walls	20	429	925	230	295
9	Floor	20	330	774	220	295
11	Floor	20	420	807	259	241
11	Walls	20	352	681	159	241
11	Drain Troughs	20	801	2,115	560	224
11	Horizontal Surfaces	20	240	691	238	224
13	Horizontal Surfaces	20	243	2,907	682	306
13	Walls	20	203	1,339	441	306
13	Floors	20	347	904	242	306
14	Floors	30	998	**5,589**	1,295	265
Total Number Of Measurements		**1,471**				

The bolded values in Table 8.2 represent values in excess of the preliminary DCGL of 5,000 dpm/100 cm^2. It is likely that the preliminary DCGL will be revised prior to initiating decommissioning activities.

It should be noted that;

- The elevated activity on the ventilation duct internals (Package 2) is likely naturally occurring.
- The elevated activity in the sub surface drains (Package #8) may be higher than reported due to attenuation and self absorption due to debris in the drain lines at the time of the characterization survey.

- The elevated activity in the condenser hot well (Package #8) may be higher than reported due to attenuation and self absorption due to debris in the bottom of the condenser at the time of the characterization survey.
- The elevated activity in the condenser expansion joint (Package #8) may be higher than reported due to attenuation and self absorption.
- The elevated measurement in the condensate system appeared to be confined to a single location.
- The elevated activity on the floor in the Fuel Building (Package #14) may be due to naturally occurring radioactivity. The Fuel Building was released previously for unrestricted use.

8.3 Removable Alpha and Beta Activity

The analysis results obtained on the smears collected to access removable alpha and beta activity as part of the characterization survey are summarized in Table 8.3. Attachment 2 contains the spreadsheets of the individual analysis results.

Table 8.3
Removable Alpha and Beta Activity

Package #	Surface or Structure	Type of Analysis	Num of Anal	Mean	Max	Standard Deviation	MDA
				dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
1	Overhead in the Turbine Building	Alpha	30	4	14	4	12
1	Overhead in the Turbine Building	Beta	30	8	90	30	85
1	Crane Catwalk	Alpha	10	1	3	2	12
1	Crane Catwalk	Beta	10	2	43	29	85
2	Floor Plugs	Alpha	40	0	5	1	13
2	Floor Plugs	Beta	40	-8	34	21	86
2	Internals of High Pressure Turbine	Alpha	20	-1	1	1	15
2	Internals of High Pressure Turbine	Beta	20	-8	34	21	84
2	Floor	Alpha	30	0	2	1	14
2	Floor	Beta	30	3	56	22	82
2	Walls Below 2 Meters	Alpha	40	-1	2	1	14

Package #	Surface or Structure	Type of Analysis	Num of Anal	Mean	Max	Standard Deviation	MDA
				dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
2	Walls Below 2 Meters	Beta	40	7	60	21	82
2	Ventilation Duct Internals	Alpha	26	1	11	4	17
2	Ventilation Duct Internals	Beta	26	12	68	24	88
3	Cooling Tower Basin Floor	Alpha	15	1	5	2	11
3	Cooling Tower Basin Floor	Beta	15	-3	43	22	85
3	Security Building Floor	Alpha	30	-1	5	1	14
3	Security Building Floor	Beta	30	-1	38	20	82
3	Rad Waste Storage Building Floor	Alpha	30	0	5	2	14
3	Rad Waste Storage Building Floor	Beta	30	1	60	24	82
3	Utility Building Floor	Alpha	30	0	5	2	14
3	Utility Building Floor	Beta	30	7	56	25	82
4	Sump 1	Alpha	10	0	2	1	11
4	Sump 1	Beta	10	-2	38	27	89
4	Sump 2	Alpha	10	0	5	2	11
4	Sump 2	Beta	10	-17	9	19	89
4	Sump 3	Alpha	10	0	2	1	11
4	Sump 3	Beta	10	-10	17	18	89
4	Sump 4	Alpha	10	1	2	1	11
4	Sump 4	Beta	10	-15	17	25	89
5	Floor Drains	Alpha	8	0	5	2	13
5	Floor Drains	Beta	8	7	26	14	86
5	Mud Drum (right side)	Alpha	10	-1	-1	0	13
5	Mud Drum (right side)	Beta	10	-14	34	26	88

Package #	Surface or Structure	Type of Analysis	Num of Anal	Mean	Max	Standard Deviation	MDA
				dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
5	Mud Drum (left side)	Alpha	10	0	5	2	13
5	Mud Drum (left side)	Beta	10	-10	9	14	86
5	Inside Boiler Tank (back side)	Alpha	10	0	2	1	13
5	Inside Boiler Tank (back side)	Beta	10	-9	21	20	88
5	Flash Tank	Alpha	5	-1	-1	0	13
5	Flash Tank	Beta	5	-13	9	17	88
5	Sump	Alpha	10	1	2	1	13
5	Sump	Beta	1-	-21	17	29	88
5	Horizontal Surfaces	Alpha	30	0	5	2	13
5	Horizontal Surfaces	Beta	30	-13	34	24	88
5	Floor	Alpha	30	0	5	2	13
5	Floor	Beta	30	-11	17	15	86
5	Drain Troughs	Alpha	20	0	5	2	13
5	Drain Troughs	Beta	20	-11	21	19	88
6	Horizontal Surfaces	Alpha	30	-1	3	2	17
6	Horizontal Surfaces	Beta	30	-5	47	24	86
6	De-aerator Tank	Alpha	10	-2	3	2	17
6	De-aerator Tank	Beta	10	-8	26	18	86
6	Exhaust Fan	Alpha	10	-2	0	1	17
6	Exhaust Fan	Beta	10	1	56	29	86
6	Floor	Alpha	30	-2	3	1	17
6	Floor	Beta	30	-7	30	21	86
6	Stack	Alpha	10	-3	-3	0	17
6	Stack	Beta	10	-7	26	23	86
7	Horizontal Surfaces	Alpha	45	0	5	2	14
7	Horizontal Surfaces	Beta	45	2	43	20	84
7	Main Steam Line	Alpha	15	1	5	2	13
7	Main Steam Line	Beta	15	-7	26	17	86
7	Air Ejectors	Alpha	10	0	5	2	13

Package #	Surface or Structure	Type of Analysis	Num of Anal	Mean	Max	Standard Deviation	MDA
				dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
7	Air Ejectors	Beta	10	-12	26	27	86
7	Walls	Alpha	45	0	5	2	14
7	Walls	Beta	45	-4	43	22	84
7	Steam Drain Trap	Alpha	10	-2	0	1	17
7	Steam Drain Trap	Beta	10	2	26	16	88
7	Floor	Alpha	45	0	5	2	14
7	Floor	Beta	45	1	60	21	84
7	Stop Valve	Alpha	10	-1	2	1	15
7	Stop Valve	Beta	10	-11	38	32	84
7	Control Valve	Alpha	10	0	5	2	15
7	Control Valve	Beta	10	-11	13	17	84
7	Hydrogen Cool	Alpha	10	0	2	1	15
7	Hydrogen Cooler	Beta	10	-12	17	18	84
7	Floor Of Electrical Room	Alpha	20	-1	3	1	15
7	Floor Of Electrical Room	Beta	20	2	47	22	84
7	Floor Of Instrument Shop	Alpha	20	-1	2	1	15
7	Floor Of Instrument Shop	Beta	20	-7	26	22	84
7	Floor Of Room East Of Rad Waste Room	Alpha	20	0	5	2	15
7	Floor Of Room East Of Rad Waste Room	Beta	20	-4	56	24	84
7	Floor Of Room South Of Rad Waste Room	Alpha	20	-1	3	2	15
7	Floor Of Room South Of Rad Waste Room	Beta	20	0	43	22	85

Package #	Surface or Structure	Type of Analysis	Num of Anal	Mean	Max	Standard Deviation	MDA
				dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
8	Condensate System	Alpha	15	-1	4	2	15
8	Condensate System	Beta	15	9	47	17	84
8	Walls	Alpha	30	-1	2	1	13
8	Walls	Beta	30	-9	21	18	86
8	Feed Water Pump	Alpha	15	-1	2	1	13
8	Feed Water Pump	Beta	15	-17	34	21	88
8	Floor Under Condenser	Alpha	20	0	5	2	13
8	Floor Under Condenser	Beta	20	-7	56	30	88
8	Floor	Alpha	30	0	5	2	13
8	Floor	Beta	30	-17	21	15	85
8	Horizontal Surfaces	Alpha	30	0	5	1	13
8	Horizontal Surfaces	Beta	30	-16	64	26	88
8	Condenser Hot Well	Alpha	22	0	5	2	13
8	Condenser Hot Well	Beta	22	-13	47	27	88
8	Upper Level Of Condenser	Alpha	20	0	5	2	13
8	Upper Level Of Condenser	Beta	20	-7	43	20	88
8	Hot Side Sump	Alpha	15	0	5	2	13
8	Hot Side Sump	Beta	15	-4	30	23	88
8	Trenches	Alpha	30	-1	0	1	16
8	Trenches	Beta	30	-5	43	22	85
8	Feed Water Heater	Alpha	15	0	2	1	13
8	Feed Water Heater	Beta	15	-18	13	16	88
8	Floor Of Maintenance Shop	Alpha	30	0	2	1	13
8	Floor Of Maintenance Shop	Beta	30	-13	9	16	88

Package #	Surface or Structure	Type of Analysis	Num of Anal	Mean	Max	Standard Deviation	MDA
				dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2	dpm/100 cm^2
8	Floor Of Maintenance Office	Alpha	20	0	5	2	13
8	Floor Of Maintenance Office	Beta	20	-11	64	29	88
8	Floor Of Laundry	Alpha	20	1	5	2	13
8	Floor Of Laundry	Beta	20	-13	17	16	88
9	Horizontal Surfaces	Alpha	20	0	5	2	13
9	Horizontal Surfaces	Beta	20	-6	34	20	86
9	Walls	Alpha	20	0	2	1	13
9	Walls	Beta	20	-9	34	23	86
9	Floor	Alpha	20	-1	2	1	13
9	Floor	Beta	20	1	30	18	86
11	Floor	Alpha	20	0	6	3	17
11	Floor	Beta	20	-5	21	15	88
11	Walls	Alpha	20	-2	0	1	17
11	Walls	Beta	20	4	47	27	88
11	Drain Troughs	Alpha	20	1	11	4	17
11	Drain Troughs	Beta	20	13	56	29	88
11	Horizontal Surfaces	Alpha	20	-1	3	2	17
11	Horizontal Surfaces	Beta	20	-7	43	24	88
13	Horizontal Surfaces	Alpha	20	0	7	2	13
13	Horizontal Surfaces	Beta	20	1	38	25	84
13	Walls	Alpha	20	0	7	2	13
13	Walls	Beta	20	-14	17	19	84
13	Floor	Alpha	20	0	5	2	13
13	Floor	Beta	20	-9	26	22	84
14	Floor	Alpha	30	1	16	4	13
14	Floor	Beta	30	-1	**94**	28	84
Total Number Of Analyses			**2,762**				

The bolded results in Table 8.3 represent values in excess of the MDA. Only two survey packages contained analyses results in excess of the MDA, packages 1 and 14. In both cases the measured activity did not exceed the MDA by more than 10 percent. The results of the analyses for removable alpha and beta activity demonstrate that essentially no removable activity was found in the areas surveyed. It should be noted that several of the surfaces and structures identified as having elevated direct beta measurement results were not surveyed for removable activity. These surface and structures include the sub surface drain lines and the condenser expansion joint.

8.4 Exposure Rate Measurements

Exposure rate measurements were collected in the Utility Building and the Rad Waste Storage Building as part of the characterization survey due to the amount of material being stored within the buildings at the time the characterization survey was performed. Table 8.4 summarizes the results of the exposure rate measurements. Attachment 3 contains the spreadsheets with the individual measurement results.

Table 8.4
Exposure Rate Measurement

Package #	Surface or Structure	Num Of Meas	Mean	Max	Standard Deviation
			uR/hr	uR/hr	uR/hr
3	Utility Building	30	6.1	12.4	1.5
3	Rad Waste Storage Building	30	5.0	5.9	0.4
Total Number Of Measurements		**60**			

The exposure rate measurements did not identify any areas containing licensed radioactive material. The slightly elevated exposure rate (12.4 uR/hr) appeared to be due to a pile of fire brick being stored within the Utility Building.

8.5 Off Site Sample Analysis results

A part of the characterization survey both biased samples from within the plant and environmental samples were collected for off site analysis. All of the samples were analyzed by gamma spectroscopy, radionuclides of interest included Ag-108m, Co-60, Cs-137, Eu-152, Eu-154, Eu-155, Mn-54, and Zn-65. Water samples were also analyzed for H-3. Selected samples, primarily those suspected of containing licensed activity were also analyzed for Ni-59, Ni-63, Sr-90, Tc-99, Cm-242, Fe-55, Pu-238, Pu-239/240, Pu-241, Th-228, Th-230, Th-232, U-234, U-235, U-238, and C-14. Table 8.5 summarizes the results of the off site sample analysis results containing activity in excess of the MDA. The results of the off site sample analysis results are provided in Attachment 4.

Table 8.5
Summary Of Off Site Samples With Detectable Activity

Sample Location	Radionuclide	Activity (pCi/g)	Error (pCi/g)	MDA (pCi/g)
Condenser Exp Joint (Turbine Building)	Co-60 Zn-65	2.35E2 1.31E0	1.55E1 5.66E-1	2.70E-1 7.62E-1
Mud Drum (Boiler Building)	Co-60	1.05E1	9.92E-1	8.76E-2
Condenser, Hot Side (Turbine Building)	Co-60	9.09E0	7.00E-1	1.48E-1
Condensate Pit (Turbine Building)	Ag-108m Co-60 Eu-155	3.18E3 2.12E0 4.42E1	2.01E2 8.70E-1 8.04E0	2.09E0 1.28E0 2.88E0
Turbine Building Sump (Turbine Building)	Co-60 Cs-137 Eu-152 Eu-155	8.25E0 8.65E-1 2.08E0 1.45E0	5.86E-1 1.28E-1 4.70E-1 2.12E-1	9.34E-2 8.98E-2 7.76E-1 2.27E-1
Drain To Sump (Fuel Building)	Ra-226	1.76E2	1.74E1	1.87E-1
Sump (Fuel Building)	Ra-226	1.18E2	1.19E1	1.39E-1
		Activity (pCi/l)	Error (pCi/l)	MDA (pCi/l)
Hydrogen Cooler (Turbine Building)	H-3	1.40E4	1.54E3	3.58E2

9.0 Conclusion

The results of the characterization survey of the Pathfinder Plant appear to demonstrates that the amount of residual activity remaining from the operation of the Pathfinder reactor is relatively low and confined to several well defined areas within the plant.

The direct beta measurements show that while residual activity in excess of the MDA was identified throughout the plant, residual activity, attributable to licensed radionuclides, in excess of the preliminary DCGL (5000 dpm/100 cm^2) was identified in only a limited number of areas. Table 9.0.1 list those surfaces and structures with residual activity in excess of the preliminary DCGL.

Table 9.0.1
Areas With Residual Radioactivity In Excess Of The Preliminary DCGL

Package #	Surface or Structure	Max Direct Beta Measurement Result
		In Excess of 5000 dpm/100 cm^2
8	Sub Surface Drain Up Stream Of Condenser	8,143
8	Sub Surface Drain Down Stream Of Condenser	38,714
8	Condensate System	12,631
8	Floor Under Condenser	8,075
8	Condenser Hot Well	5,849
8	Condenser Expansion Joint	12,757

The exposure rate measurements and the analyses for removable alpha and beta activity did not identify any residual activity attributable to licensed radionuclides.

The baised sample analysis results collected within the Pathfinder Plant identified residual activity attributable to licensed radionuclides in several locations. Table 9.0.2 lists those areas from which samples were collected that resulted in detectable activity attributable to licensed radionuclides in excess of MDA.

Table 9.0.2
Biased Sample Analysis Results In Excess Of MDA

Sample Location	Radionuclide	Activity (pCi/g)
Condenser Exp Joint (Turbine Building)	Co-60 Zn-65	2.35E2 1.31E0
Mud Drum (Boiler Building)	Co-60	1.05E1
Condenser, Hot Side (Turbine Building)	Co-60	9.09E0
Condensate Pit (Turbine Building)	Ag-108m Co-60 Eu-155	3.18E3 2.12E0 4.42E1
Turbine Building Sump (Turbine Building)	Co-60 Cs-137 Eu-152 Eu-155	8.25E0 8.65E-1 2.08E0 1.45E0
Sample Location	Radionuclide	Activity (pCi/l)
Hydrogen Cooler (Turbine Building)	H-3	1.40E4

As Table 9.0.2 shows the radionuclides of interest during future D&D activities may include: H-3, Co-60, Zn-65, Ag-108m, Cs-137, Eu-152, and Eu-155. This list of radionuclides should be reviewed prior to performing a final status survey in order to account for additional radioactive decay (Zn-65) and the effects of any remedial activities.

The environmental samples collected in the environs surrounding the Pathfinder Plant did not identify any residual radioactivity attributable to licensed radionuclides in excess of MDA.

10.0 References

10.1 Characterization Survey Plan for the Pathfinder Plant in Sioux Falls, South Dakota, Revision 0, October 2003.

10.2 NUREG-1575, Multi-Agency Radiation and Site Investigation Manual (MARSSIM), August 2000.

10.3 NUREG-1757, Consolidated NMSS Decommissioning Guidance

10.3.1 NUREG-1757, Volume 1, Decommissioning Process for material Licensees, Final, September 2002

10.3.2 NUREG-1757, Volume 2, Characterization, Survey, and Determination of Radiological Criteria, Draft for Comment, September 2002.

10.4 PNL-4326, Topical Report, Residual Radionuclide Distribution and Inventory At The Pathfinder Generating Plant, June 1982.

10.5 NUREG/CR-3474, Long Lived Activation Products in Reactor Material, August 1984

Appendix E

Pathfinder Quality Assurance Project Plan

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PATHFINDER DECOMMISSIONING PROJECT

QUALITY ASSURANCE PROJECT PLAN

REV. 0

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QUALITY ASSURANCE PROJECT PLAN

1. POLICY STATEMENT

Xcel Energy is responsible for the safe decommissioning of the Pathfinder site. Decontamination and decommissioning activities at the Pathfinder site are subject to the general requirements of the Utility Engineering Quality Assurance Manual (QAM) and the specific requirements of the Pathfinder Decommissioning Quality Assurance Project Plan (QAPP). In the event conflicts exist between requirements of the QAM and the QAPP, the requirements of the QAPP will prevail. The quality assurance program, as applied to activities shall comply with and be responsive to applicable regulatory requirements and applicable industry codes and standards. These activities are for the protection of the health and safety of the public and project personnel, and for adherence to regulations and commitments made to the Nuclear Regulatory Commission, including the control of personnel exposure to radiation, control of radioactive material and contamination, and radwaste shipment.

Project procedures shall provide for compliance with appropriate regulatory, statutory, license, and industry requirements. Specific quality assurance requirements and organizational responsibilities for implementation of these requirements shall be specified.

Compliance with this program and provisions of project procedures is mandatory for personnel with respect to Pathfinder decommissioning activities, which may affect quality or the health and safety of project personnel or the general public. Personnel shall, therefore, be familiar with the requirements and responsibilities of the program that are applicable to their individual activities and interfaces.

2. INTRODUCTION

This project quality assurance program is structured to comply with the appropriate regulatory requirements of NUREG 1757, Consolidated NMSS Decommissioning Guidance, and is implemented to assure that surveying, dismantling, packaging, and shipping activities are conducted in a controlled manner designed to assure quality and to protect the health and safety of both project workers and the public.

3. ORGANIZATION

The decommissioning organization for the Pathfinder Decontamination & Decommissioning (D&D) is shown in Figure 1.1. The Director - Xcel Nuclear Asset Management has the management authority for the safe dismantlement and decommissioning of the Pathfinder site. He has overall responsibility for implementing this Quality Assurance Project Plan. The key decommissioning staff members perform the functions described in the following subsections.

3.1 PROJECT MANAGER

Directly responsible to the Director - Xcel Nuclear Asset Management, the Project Manager coordinates and oversees all decommissioning activities. This person directs subordinates and support contractors to ensure radiological and industrial safety, compliance with regulatory and procedural requirements, and cost-effectiveness of the decommissioning project. The Project Manager implements the Quality Assurance Project Plan and is the contact point for contested items from QA and corporate industrial safety. This person provides necessary liaison with regulatory agencies and utility management.

The Project Manager may perform the duties of the Project Engineer or delegate these duties as necessary.

3.1.1 PROJECT ENGINEER

This person supervises engineering support and or construction personnel and assists the construction superintendent in developing detailed work procedures. This person arranges the writing of specifications for special equipment, tools, and services that must be procured or fabricated. The Project Engineer prepares reports requested by the decommissioning Project Manager and is responsible for licensing activities.

3.1.2 RADIATION SAFETY OFFICER (RSO)

The Radiation Safety Officer is responsible for ensuring compliance with radiation work procedures. This individual is responsible for directing the activities of the Radiation Protection Specialist(s). The RSO oversees decommissioning activities, recording of on the job radiation dose information and operation of the plant laboratory facilities including sampling and analysis. He supervises the Radioactive Waste Shipping Specialist in all radioactive material shipments.

This individual is also qualified to perform the duties of the Radiation Protection Specialist and may do as the workload dictates.

The RSO may hire additional contractor personnel, or Xcel Energy nuclear plant and/or Xcel Energy corporate personnel. They may assist with job supervision during peak times, or as operations dictate.

3.1.3 CONSTRUCTION SUPERINTENDENT

The Construction Superintendent is responsible for carrying out the actual decommissioning work during a shift, and this individual supervises Xcel Energy crew leaders and craft supervisors. This person reports to the Project Manager. As this person supervises the day-to-day performance of the shift, this person recommends changes in procedures and schedules to improve the safety and/or cost effectiveness of the project. This person also is responsible for directing and supervising work performed by dismantling subcontractors.

3.1.4 PROJECT CONTROLS SUPERVISOR

The Project Controls supervisor is responsible for establishing cost controls, managing contracts, and preparing and maintaining project schedules.

3.1.5 ADMINISTRATIVE CONTROLS SUPERVISOR

The Administrative Controls Supervisor is responsible for issuing controlled documents and the retention of quality records.

3.2 QUALITY ASSURANCE MANAGER

The Quality Assurance Manager reports to the Director - Xcel Nuclear Asset Management and is responsible for assessing implementation of the quality assurance plan for decommissioning. He provides consultation and advice to the Project Manager regarding implementation of the Quality Assurance Program. This person manages the independent assessment function, maintains audit and surveillance records, and verifies that established project procedures are followed for quality-related activities.

3.3 PLANT MANAGER

The Plant Manager is responsible for industrial security and industrial safety at the Pathfinder site.

4. PATHFINDER QUALITY ASSURANCE PROGRAM

GENERAL REQUIREMENTS

A) The project quality assurance program shall be documented by written procedures and carried out throughout the decommissioning project in accordance with those procedures.

B) The program shall apply to radiological protection and survey activities and the packaging and shipping of radioactive waste.

C) The program shall provide control over activities affecting quality or the health and safety of project personnel and the public.

D) Activities affecting quality shall be accomplished under suitable controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been satisfied.

E) The program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of satisfactory implementation.

F) The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained.

G) The adequacy and status of the program shall be regularly reviewed.

H) Management of other organizations participating in the program shall regularly review the status and adequacy of the part of the program which they are implementing.

4.1 GENERAL DESCRIPTION

A) The Pathfinder Decommissioning Project Quality Assurance Program has been established to govern those activities that may affect the quality of the project, including the health and safety of the public as well as the project personnel.

B) The project quality assurance program shall utilize the following documents to meet its objectives.

1. Pathfinder Decommissioning Quality Assurance Project Plan (QAPP).
2. The Utility Engineering Quality Assurance Manual (QAM).
3. Required procedures at the project implementing level.

The QAPP contains mandatory requirements that must be met for quality-related activities, but is considered as a guidance document for project work.

4.2 PATHFINDER DECOMMISSIONING QUALITY ASSURANCE PROJECT PLAN

A) The QAPP shall describe in general how compliance with appropriate quality and safety requirements is accomplished.

B) The QAPP shall be issued under the authority of the Director - Xcel Nuclear Asset Management.

C) All changes to the QAPP shall be approved by the Director - Xcel Nuclear Asset Management.

4.3 QUALITY ASSURANCE TRAINING

A) Personnel responsible for performing activities affecting quality or the health and safety of project personnel or the general public are instructed as to the purpose, scope, and implementation of application controlling procedures.

B) Personnel performing such activities are trained and qualified, as appropriate, in principles and techniques of the activity being performed.

C) The scope, the objective, and the method of implementing the training programs are documented.

D) Methods are provided for documenting training sessions describing content, attendance, data of attendance, and the results of the training session, as appropriate.

5. DESIGN CONTROL

GENERAL REQUIREMENTS

Modifications are designed in accordance with the requirements of the Utility Engineering QAM Section 4 - Design Control.

6. PROCUREMENT DOCUMENT CONTROL

Procurement documents are prepared in accordance with the requirements of the Utility Engineering QAM Section 6 - Purchasing. The responsible project organization shall assure those applicable regulatory requirements, design bases, and other requirements that are necessary to assure adequate quality are suitably included or referenced in the procurement document.

7. PROCEDURES AND DRAWINGS

GENERAL REQUIREMENTS

A) Procedures and drawings of a type appropriate to the circumstances shall be provided for the control and performance of activities that are important to quality, health, and safety.

B) Procedures and drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

7.1 PROCEDURES

A) Typical working level procedures include the following, as appropriate.

Calibration procedures
Radiation protection procedures
Radioactive material packaging and shipment procedures
Audit or surveillance procedures
Administrative control procedures (e.g. Corrective Action, Document Control, Records Retention, Audit, etc.)

B) Procedures shall provide specific controls and instructions for performing activities affecting quality or the health and safety of project personnel or the general public.

Procedures shall be reviewed by technically competent persons other than the preparer and approved by project management.

Contractor and third party procedures shall receive independent technical review and project management approval.

7.2 DRAWINGS

Drawings are controlled in accordance with the requirements of the Utility Engineering QAM Section 5 F - Document and Data Control.

8. DOCUMENT CONTROL

GENERAL REQUIREMENTS

A) Measures shall be established to control the issuance of documents, such as procedures and drawings, including changes thereto, which prescribe activities affecting quality.

B) These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel, and are distributed to and used at the location where the prescribed activity is performed.

C) Changes to documents shall be reviewed and approved by the same organization that performed the original review and approval or by another designated responsible organization.

8.1 PROCEDURE CONTROL

Required procedures shall be controlled to assure that current copies are made available to personnel performing the prescribed activities. Required procedures shall be reviewed by a technically competent person other than the preparer and shall be approved by a member of the project management staff.

8.2 RADIOACTIVE SHIPMENT PACKAGE DOCUMENTS

All documents related to a specific shipping package for radioactive material shall be controlled by appropriate procedures. All significant changes to such documents shall be similarly controlled.

9. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

GENERAL REQUIREMENTS

Purchased materials, equipment, and services are controlled in accordance with the requirements of the Utility Engineering QAM Section 6 - Purchasing.

9.1 RECEIPT INSPECTION

A) Commensurate with potential adverse impacts on quality or health and safety, material and equipment shall be inspected upon receipt at the plant site prior to use or storage to determine that procurement requirements are satisfied.

B) Material, parts, and components that are to be utilized to fulfill a 10 CFR 71 related function or used for shipment of radioactive materials shall be inspected upon receipt to assure that associated procurement document provisions have been satisfied. Measures shall be established for identifying nonconforming material, parts and components.

10. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS AND COMPONENTS

Materials parts and components are identified and controlled in accordance with the requirements of the Utility Engineering QAM Section 8 - Product Identification and Traceability.

Site procedures may be developed to supplement the Utility Engineering Quality Assurance Manual requirements as necessary.

11. CONTROL OF SPECIAL PROCESSES

GENERAL REQUIREMENTS

Measures shall be established to assure that special processes, including welding, and nondestructive examination are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

11.1 WELDING PROCEDURES

Welding of critical lifting and rigging equipment shall be performed in accordance with qualified procedures. Such procedures shall be qualified in accordance with applicable codes and standards and shall be reviewed to assure their technical adequacy.

11.2 WELDER QUALIFICATION

Measures shall be established that assure welding of critical lifting and rigging equipment is performed by qualified personnel.

11.3 NDE PROCEDURES

Nondestructive examinations (NDE) of critical lifting equipment shall be performed in accordance with procedures formulated in accordance with applicable codes and standards and shall be reviewed to assure their technical adequacy.

11.4 NDE PERSONNEL QUALIFICATION

Measures shall be established that assure nondestructive examination (NDE) are performed by personnel qualified in accordance with applicable codes and standards.

12. INSPECTION

GENERAL REQUIREMENTS

A) Measures shall be established for inspection of appropriate activities to verify conformance with the documented procedures and drawings for accomplishing the activity.

B) If mandatory inspection hold points, which require witnessing or inspection and beyond which work shall not proceed without prior consent are required, the specific hold points shall be indicated in appropriate document.

12.1 TECHNICAL SERVICES

Measures shall be established which assure that activities associated with technical services (such as surveillance testing, instrument calibration, laboratory services, etc.) are inspected by qualified personnel when determined appropriate by quality or other qualified personnel.

12.2 RADIOACTIVE MATERIAL PACKAGES

Measures shall be established which assure that packages utilized to ship licensed radioactive material offsite are inspected in accordance with the applicable provisions of 10 CFR 71.

12.3 INSPECTION PROCEDURES

Required inspections shall be performed in accordance with appropriate procedures. Such procedures shall contain a description of objectives, acceptance criteria and prerequisites for performing the inspections. These procedures shall also specify any special equipment or calibrations required to conduct the inspection.

12.4 PERSONNEL QUALIFICATION

A) Personnel performing required inspections shall be qualified. Required inspections shall not be performed by individuals who performed the inspected activity or directly supervised the inspected activity.

B) Personnel performing inspections required by sections required by sections 12.2 and 12.3 shall be qualified based upon experience and training in inspection methods.

12.5 SELF-CHECKING AND PEER VERIFICATION

Workers may use self-checking and peer verification techniques to assure activities meet quality requirements. Self-checking and peer verification may be used in addition to, but shall not be used as a substitute for inspection when an inspection is required.

13. TEST CONTROL

GENERAL REQUIREMENTS

Measures shall be established to assure that tests necessary to assure quality or health and safety are controlled and accomplished in accordance with quality requirements. Such tests include verification of lifting capacity of cranes.

14. CONTROL OF MEASURING AND TEST EQUIPMENT

GENERAL REQUIREMENTS

Measures shall be established to assure that tools, gauges, instruments and other measuring and testing devices used in activities important to health and safety are properly controlled, calibrated and adjusted at specified periods to maintain accuracy within necessary limits.

15. HANDLING, SHIPPING, AND STORAGE

GENERAL REQUIREMENTS

Handling, storage and shipping of general items is controlled in accordance with the requirements of the Utility Engineering QAM Section 15, Handling, Storage, Packaging, Preservation and Delivery.

15.1 RADIOACTIVE MATERIAL STORAGE

A) Areas shall be provided for storage of radioactive material which assure physical protection, as low as reasonably achievable radiation exposure to personnel, control of stored material, and containment of radioactive material, and containment of radioactive material as appropriate.

B) Handling, storage, and shipment of radioactive material shall be controlled based upon the following criteria.

1. Established safety restrictions concerning the handling, storage, and shipping of packages for radioactive material shall be followed.

2. Shipments shall not be made unless all tests, certifications, acceptances, and final inspections have been completed.

3. Procedures shall be provided for handling, storage and shipping operations.

15.2 RADIOACTIVE MATERIAL SHIPPING AND PACKAGING

Shipping and packaging documents for radioactive materials shall be consistent with the applicable requirements of 10 CFR 71.

16. INSPECTION, TEST, AND OPERATING STATUS

The operating status of quality-related equipment is indicated by tagging or other specified means to prevent inadvertent use. The status of inspections or test performed on individual items is clearly indicated by markings and/or logging to prevent inadvertent by passing of such inspections and tests.

17. NONCONFORMING MATERIALS, PARTS OR COMPONENTS

GENERAL REQUIREMENTS

A) Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or release for shipment. These measures shall include, as appropriate, procedures for identification, documentation, segregations, disposition and notification to affected organizations.

B) Nonconformance items shall be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

18. CORRECTIVE ACTION

GENERAL REQUIREMENTS

A) Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, discrepancies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

B) For significant conditions adverse to quality, the cause of the condition shall be determined and corrective action taken to preclude recurrence. In these instances, the condition, cause and corrective action taken shall be documented and reported to appropriate levels of management.

19. QUALITY ASSURANCE RECORDS

GENERAL REQUIREMENTS

A) Sufficient records shall be maintained to furnish evidence of activities affecting quality. These records shall be consistent with the requirements of Section 3, Record Keeping, of Volume 3, Financial Assurance Record Keeping and Timeliness, of NUREG-1757.

B) Records shall be identifiable and retrievable.

Requirements shall be established concerning record retention, such as duration, location, and assigned responsibility. Such requirements shall be consistent with the potential impact on quality, health and safety of public, safety of project personnel, and applicable regulations.

20. AUDITS

GENERAL REQUIREMENTS

A system of planned audits shall be carried out to verify compliance with appropriate requirements of the Project Qualify Assurance Program and to determine the effectiveness of the program. The audits shall be performed in accordance with written procedures or checklists by appropriately trained personnel no having direct responsibility in the areas being audited. Audit results shall be documented and reviewed by management having responsibility in the area audited. Follow-up action, including re-audit of discrepant areas, shall be taken where indicated.

20.1 AUDIT REPORTS

A) Reports of the results of each audit shall be prepared. These reports shall include a description of the area audited, identification of individuals responsible for implementation of the audited provisions and for performance of the audit, identification of discrepant areas, and recommended corrective action as appropriated.

B) Audit reports shall be distributed to the appropriate management level and to those individuals responsible for implementation of audited provisions.

20.2 CORRECTIVE ACTION

Measures shall be established which assure that discrepancies identified by audits or other means are resolved. These measures shall include notification of the manager responsible for the discrepancy, recommended corrective action, and verification of satisfactory resolution. Discrepancies shall be resolved by the manager responsible for the discrepancy. Management shall resolve disputed discrepancies.

Figure 1

Pathfinder Quality Project Organization